Limerick D3 Analysis Evaluation Open Item Summary Table							
Item No.	**Source Branch**	**Location in Application**	**Issue Description** *(From NRC Staff)*	**Licensee Response**	**Status** *(From NRC Staff)*	**Audit, RAI, or RCI Number** *(From NRC Staff)*	**Licensee Supplement** *(if any)*
1	EICB		**Bold / Underlined / Italics Notation**: Section 3.46 (P. 3-71) of the topical report includes the following statement: "Bold, underlined, and italicized texts for in the Device Tag # column of the table indicates instruments that need to be shared from PPS." This is inconsistent with the note in Section 4.6 (P. 4-6) which provides a different definition for text that is formatted in this way as follows: "Note: The use of bold underline italic text below indicates diverse features that are required to be implemented in diverse protection system (DPS)."	The above statement refers to display instruments that need to be shared from PPS and implemented in DPS. The statement in Section 3.46 will be revised in the D3 analysis to, "Bold, underlined, and italicized texts in the Device Tag # column of the table indicates additional instrumentation that need to be shared from PPS for diverse indication at the DPS". Section 4.6 are diverse controls that are required to be implemented in DPS. The statement in Section 4.6 will be revised in the D3 analysis to, "The use of bold underline italic text below indicates diverse controls that are required to be implemented in DPS."	CONFIRM~~Open~~		
2	EICB		**DPS Function Clarification:** The following indications and controls are identified in the topical report as being both required DPS functions and as non-DPS functions in the sections indicated: • Core Spray Loop Flow – Identified as non-DPS in Sections 3.33.5 & 3.34.5 Identified as DPS function in Sections 3.14.5, 3.15.5 & 3.16.5 • Digital Electro-Hydraulic Control System (DEHC) – Identified as non-DPS in Sections 3.1.5, 3.3.5, 3.5.5, 3.6.5, 3.9.5, 3.11.5, 3.12.5, 3.15.5, 3.17.5, 3.18.5, 3.19.5, 3.21.5, 3.22.5, 3.26.5, 3.27.5, 3.28.5, 3.31.5, Identified as DPS function in Section 3.13.5, • Suppression Pool Water Level – Identified as non-DPS in Section 3.33.5 Identified as DPS function in Section 3.34.5	Sections 3.33.5 and 3.34.5 will be revised to indicate a DPS function.] [Sections 3.13.5 will be revised to indicate DEHC as a non-DPS function. Section 3.33.5 will be revised to indicate a DPS function.	CONFIRM~~OPEN~~		

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3	EICB		**Component Interface Module:** Port Z of the component interface module (CIM) is shown to be connected to the distributed control system (DCS) / redundant reactivity control system (RRCS) as an optional command control input to the CIM. This is presumed to be the input from a diverse protection system. This would give non-safety DCS or RRCS commands higher priority than the command signals originating from the safety related PPS integrated logic processors. Is Limerick planning to exercise this option or not? Why is this listed as an option? Has the decision to include automatic diverse actuation (DAS) functions through the CIM not yet been made? If DAS priority is used, then we would need justification for allowing nonsafety-related system commands to override the commands from the safety related PPS system. 6/16/22: This response explains that DAS can only initiate safety functions to safe state over the Z port of the CIM. The response also cites ISG 4 allowances for this approach, but this justification is still not included in the analysis. Is there going to be an added justification in the next revision to the analysis?	Is Limerick planning to exercise this option or not? RESPONSE: Yes. Why is this listed as an option? RESPONSE: Not every field component is controlled by the DPS/RRCS (i.e., DCS), only a subset is. Has the decision to include automatic diverse actuation (DAS) functions through the CIM not yet been made? RESPONSE: The design decision has been made; the Z-port will be used to support automatic DPS functions. If DAS priority is used, then we would need justification for allowing NSR system commands to override the commands from the safety related PPS system. RESPONSE: As stated in NRC Interim Staff Guidance DI&C-ISG-04, Revision 1, Section 2, "safety-related commands that direct a component to a safe state must always have the highest priority and must override all other commands". In the case of the PPS and DPS, the DPS interface at the Z-port only receives commands to the safe-state. In other words, the DPS overrides the PPS command when the DPS issues a command to the safe-state. The DPS only provides a redundant means to initiate the safety function via the Z-port. DPS does not override the PPS safety function.	CONFIRM~~OPEN~~		

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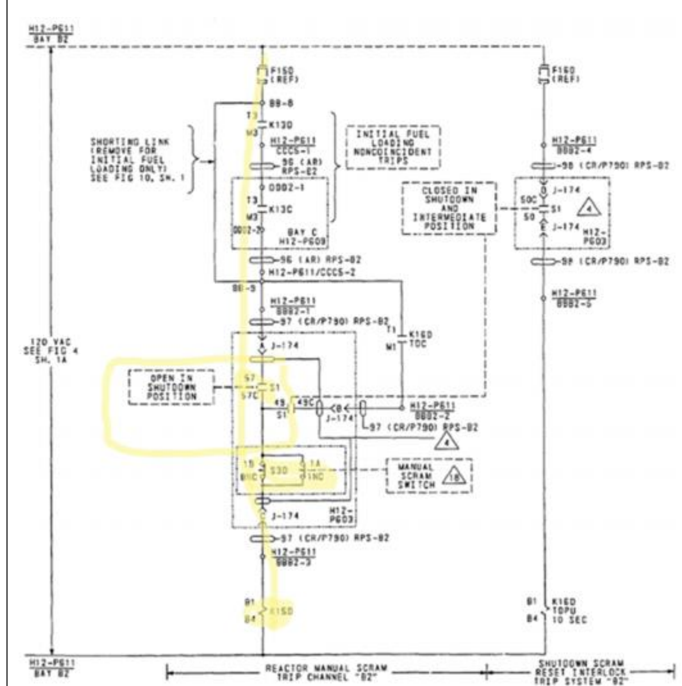
4	EICB	D3 Section 2.1	**PPS Architecture:** The PPS architectural diagram (Figure 2-1) provided in the topical report shows signal inputs being provided to the DCS through isolators however, these inputs are labeled as "(optional)." To evaluate the level of independence established between the PPS and the DCS, the NRC will need to know if these options are going to be exercised and the degree of isolation that would be established by these isolators. It is also unclear if these PPS to DCS interfaces are strictly analog signals or if digital communications links are being used. 6/16/22: We understand the response, but will the added clarification include information on which signals share signals with PPS and have qualified isolation devices. We would still like to verify that controls and functions which share signals are properly isolated and the analysis does not currently have design information to support this	Optional means not all PPS inputs are shared with the Non-safety DCS (i.e., DPS/RRCS). When PPS and DCS share an input, a qualified isolator will be installed to prevent any failures occurring on the DPS input side from adversely affecting the PPS input side. The D3 analysis will be updated to clarify this point.	OPEN		

5	EICB		**Mode Selector Switch:** The mode selector switch input to the PPS is shown as an input to the Local Coincidence Logic module in each division of the PPS. This indicates that the reactor scram function of the Mode Selector Switch may be dependent on PPS functionality to initiate a reactor scram.	6/1/22: The UFSAR only refers to the Mode selector switch scram function as a backup to the RPS manual controls (see Section 7.2.2.1.2.3.1.17, "RPS – IEEE 279 (1971), Paragraph 4.17"; not a diverse backup. Therefore, there is no reduction in diversity. As shown in Figure 1, the existing LGS configuration of the Mode Selector Switch and the RPS manual actuation logic share a significant amount of circuitry that feeds into the final actuation circuitry. Although the Reactor Mode Switch is a separate input into the scram circuit, it is not a diverse scram signal. **[** [redacted] **]]** there is also the PPS Manual Scram (reactor scram push buttons) that interfaces directly to the RPS TUs, bypassing all PPS software as discussed in section 2.1. Figure 1 - Existing LGS Reactor Mode Switch and RPS Manual Scram Circuitry 	CONFIRM~~OPEN~~		

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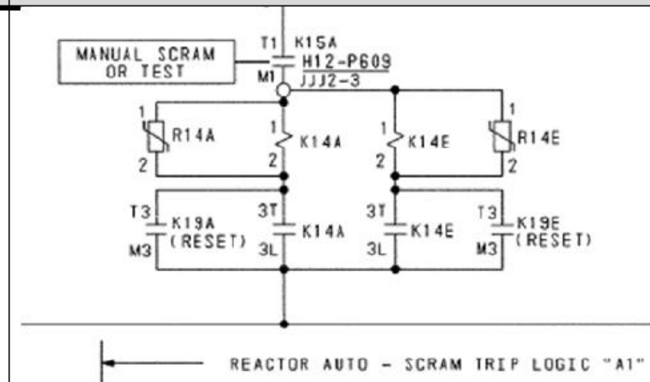
			6/16/22: Regarding the Mode Switch, the modification introduces a new dependency for mode switch functions on Tricon software. This is a dependency that wasn't present before so we are not sure that the assertion that there is no reduction in diversity is true. It seems like there is no reduction in credited diversity and that may be OK				
6	EICB		**Component Interface Module:** On page 2-7 of the topical report, the following statement is made: "The CIM is capable of performing the following functions. Not all the CIM functions listed are likely to be used in the Limerick implementation of the PPS. The design documentation will specify the CIM functions that will be employed: • Full Stroke Lock-out • Command Latch • Thermal Overload Block • Anti-hammering Logic • Anti-coincidence Logic • Discrepancy Detection" It will be necessary for the NRC staff to understand all aspects of CIM functionality in order to complete its evaluation. Therefore, the licensee is requested to identify which of these will be used in the PPS design. 6/16/22: Can this response be included as a clarification in the revision to the analysis?	5/31/22: The D3 Analysis will be updated with the following text, "None of the listed functions except for the Anti-coincidence Logic will be used for the LGS DMP. [[[redacted]]]	CONFIRM~~OPEN~~		

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7	EICB		**PPS Architecture - HARP Modules:** The architecture diagram includes devices labeled as HARP which are between the CIM modules and the field components. The abbreviation HARP is not defined in the analysis and there is no discussion of these components in the topical report. What are these HARP components and what function do they perform? Since these components are in line with the actuated components, do they operate independently from the PPS, or do they rely on PPS functionality to perform their required field device actuation function? Similarly, if the HARP modules require control power from the PPS or provide motive power to actuated devices, describe whether this power is independent of or a part of the control power or motive power in the PPS cabinets.	The High Amperage Relay Panel (HARP) assembly are interposing solid state relays utilized in conjunction with the Component Interface Module (CIM) outputs for field component interfaces requiring high amperage output capability. There is no programable digital device as part of the HARP. The HARP only responds to the CIM outputs independently of the PPS. The D3 analysis will be revised with this response. 5/31/22: [[[redacted]]]	CONFIRM~~OPEN~~		
8	EICB	D3 Section 3, and 6	**Diverse Protection System Design:** There are several diverse indications such as reactor pressure vessel Level Narrow Range that are not designated as being required DPS indications in the Section 3 coping analyses (i.e., not bold, underlined or italics) however, the diverse Indications summary Table 3-2 includes these as required indications. This is confusing because the table indicates these are diverse indications which are required, however, the references to these indications do not identify them as being required DPS indications. It is not clear if this means that these indications are provided by a non-DPS systems and that they are required, or if this means that these indications are not required. If they are provided by non-DPS systems, then it will be necessary to identify the systems performing these functions in order to access if required diversity exists between the PPS and these other systems.	The diverse system that provides the display will be identified for each analysis. For the example cited, RPV Level Narrow Range, it is provided by the diverse DFWLCS using narrow range level transmitters LT-042*N004A(B,C,D) as indicated in Table 3.2. However, in Table 3.2, a diverse RPV Water Level-Low Alarm (Level 3) is needed for EOP Entry Conditions. Thus, the Narrow Range Water Level transmitters LT-042-*N080A(B,C,D) are needed for DPS to generate this alarm. Another example where a non-safety diverse display is taken credit for in the analyses is RPV Pressure (WR), PI-042-*R605. In Table 3.2. However, DPS RVP Pressure (WR) signals based upon PT-042-*N078A(B,C,D) are needed for DPS CS initiation logic and RRCS initiation logic. The D3 analysis will be revised with this response. 6/1/22: The scope of Section 6 will be broadened to address the diversity attributes of all systems that are credited to cope with a PPS CCF. A new Table 6-1 in that section titled, "Diversity Attributes of Control and Monitoring System" will be added summarizing the diverse attributes of the credited non-safety systems used in the D3 analysis.	OPEN		

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9	EICB		**Coping Analysis:** Table 3-2 indications RPV Water Level Wide Range, Reactor Water Level Low Alarm (Level3), and DEHC, do not include references to events 15.7.1.1, 15.7.1.2, or 15.7.1.3. These three events refer to Event 15.2.5 for operator actions required. Since these are required indications for Event 15.2.5, shouldn't they also be required for events that refer to Event 15.2.5?	For Event 15.2.5 Loss of Condenser Vacuum, DEHC is a diverse non-safety system and thus it will be revised to indicate it is not a DPS function. In Table 3-2, RPV Water Level Wide Range, Reactor Water Level Low Alarm (Level 3) will be revised to indicate events 15.7.1.1, 15.7.1.2, or 15.7.1.3. The D3 analysis will be revised with this response.	CONFIRM~~OPEN~~		
10	EICB		**Coping Analysis:** Table 3-2 includes indication of RPV Pressure (WR) which is designated as a DPS indication, and which refers to event 15.6.5. However, the analysis of event 15.6.5 in Section 3.34 of the topical report does not identify Reactor Pressure as being a required DPS indication in the summary of diverse features list. 6/16/22: We are not sure we understand what you plan to change for this. Please walk us through the changes you plan to make to address this.	6/16/22: Section 3.34 will be revised to indicate reactor pressure (WR) as a DPS indication. The Table 3-2 row for RPV Pressure (WR) [PT-042-*N078A(B,C,D)] is required for automatic initiation of Core Spray, as shown in Table 3-1, and RRCS initiation logic. Table 3-2 is reserved to list the required diverse indications and therefore this row will be moved to Table 3-3. Inputs required for DPS/RRCS automatic controls will be defined in a new Table 3-3. The D3 analysis will be revised with this response.	OPEN		
11	EICB		**Coping Analysis:** Table 3-2 includes an indication of Main Steam Line Pressure which is designated as a DPS indication, and which refers to events 15.1.3 and 15.6.5. However, the analyses of these two events in Sections 3.4, 3.33 and 3.34 of the topical report do not identify Main Steam Line Pressure as being a required DPS indication in the summary of diverse features lists. 6/16/22: We are not sure we understand what you plan to change for this. Please walk us through the changes you plan to make to address this.	6/16~~5~~/22 update. In Sections 3.4.5, the main steam line pressure ~~signal credited for in the analysis refers to an existing diverse non-safety pressure indictor on Panel *0-C653~~indication will be revised to indicate a DPS signal. In Sections 3.33 (LOCA Inside Containment), the automatic MSIV closure is based upon RPV Level 1 and thus main steam line pressure is not taken credit for this function. In Section 3.34.5, the DPS main steam line pressure signals will be added. These signals are shared from PPS, and will be used for the DPS automatic MSIV closure logic. Therefore, the DPS main steam line pressure signals will be moved to Table 3-3. The D3 analysis will be revised with this response.	OPEN		

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12	EICB		**Coping Analysis – DPS Design:** Table 3-2 includes an indication of core spray (CS) Loop-A Flow which is designated as a DPS indication, and which refers to event 15.6.5. However, the analyses of event 15.6.5 in Section 3.33 of the topical report does not identify core spray Loop-A Flow as being a required DPS indication in the summary of diverse features lists. Also, events 15.2.6, 15.2.7, and 15.2.9 identify the required indication as "CS loop flow" while Table 3-2 identifies the required indication as "core spray Loop-A Flow." Please clarify if these indications are the same, and if so, explain why different labels are assigned to them. 6/16/22: We are not sure we understand what you plan to change for this. Please walk us through the changes you plan to make to address this.	6/16/22: Event 15.6.5 in Section 3.33 will be revised to indicate "Core Spray Loop-A flow" as a DPS indication and Events 15.2.6, 15.2.7, and 15.2.9 will be revised to indicate "Core Spray Loop-A flow as a DPS indication". The D3 analysis will be revised with this response.	OPEN		
13	EICB		**Coping Analysis:** Table 3-2 includes an indication of Core Spray Loop-A Pressure which is designated as a DPS indication, and which refers to event 15.6.5. However, the analyses of event 15.6.5 in Section 3.33 of the topical report does not identify Core Spray Loop-A Pressure as being a required DPS indication in the summary of diverse features lists. Also, events 15.2.6, 15.2.7, and 15.2.9 identify the required indication as "CS pump discharge pressure" while Table 3-2 identifies the required indication as "Core Spray Loop-A Pressure." Please clarify if these indications are the same, and if so, explain why different labels are assigned to them. 6/16/22: We are not sure we understand what you plan to change for this. Please walk us through the changes you plan to make to address this.	6/16/22: Event 15.6.5 in Section 3.33 will be revised to indicate "Core Spray Loop-A pressure" as a DPS indication. 6/16/22: Events 15.2.6, 15.2.7, and 15.2.9 will be revised to indicate "Core Spray Loop-A pressure" as a DPS indication. The D3 analysis will be revised with this response.	OPEN		

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14	EICB	D3 Section 3, 4, and 6	**Coping Analysis – Diverse System Functions:** There are several controls and indications that are credited in the Section 3 coping analysis but are not listed or analyzed in Section 4, "BTP 7-19 Displays and Controls." The NRC staff will need to know what systems perform each of these control and indication functions and will need to have design information on these systems to support a determination that these systems are independent of the PPS such that a common cause failure (CCF) in the PPS will not impact the ability of the subject systems to perform these credited functions. 6/16/22: This response basically says that the information we need is not in Section 4 of the analysis, but we still need the information in order to complete our evaluation. We understand that Section 4 addresses position 4 displays and controls, but we will still need to determine that all displays and controls that are credited in the Section 3 analyses will remain functional in the presence of the postulated software CCF. Therefore, we would still request that the licensee provide information on the systems that perform these functions so that we can assess their independence from the effects of PPS software CCF	Section 4 of the D3 Analysis is entitled "BTP 7-19 **POSITION 4** DISPLAYS AND CONTROLS" (bold and italics added for emphasis). This is a separate analysis based on different criteria in BTP 7-19 than the coping analysis criteria in Section 3. The criteria for Section 4 are to maintain the plant Critical Safety Functions using diverse and independent displays and controls. Therefore, not all indications and controls needed for Section 3 are necessary for Section 4. 7/1/22: Table 6-1 in Rev. 2 of the D3 Analysis will provide the requested information for both Sections 3 and 4.	OPEN		
15	EICB		**Coping Analysis:** There appears to be an error in the first sentence of Section 3.16.6 conclusion statement as follows: "For the postulated event of Loss of Shutdown Cooling Operation, existing diverse displays and controls are sufficient to mitigate the postulated event, concurrent with a postulated CCF of PPS, sufficient automatic control functions, indications that are independent of the PPS, and operator actions, are/will be available to mitigate the event." The first clause implies that controls and indications are sufficient to mitigate the postulated event with no required manual operator actions, but the clause is then repeated with the addition of operator actions.	The D3 analysis will be revised to state, "For the postulated event of Loss of Shutdown Cooling Operation, existing diverse displays with the addition of DPS diverse displays and controls identified in 3.16.5, and operator actions, are sufficient to mitigate the event concurrent with a postulated CCF of PPS." The D3 analysis will be revised with this response.	CONFIRM~~OPEN~~		

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16	SNSB		**Limerick UFSAR:** In the description of Chapter 15 events in the UFSAR, there are some noted differences between Unit 1 and Unit 2 (Sections 15.1.3 and 15.2.1 are examples). These differences should be addressed in the D3 CCF coping analysis, either generically, or for each specific event as appropriate.	There are no differences between LGS Units 1&2 in terms of system response, functions, and accident sequences and thus the analyses are applicable for both units. In the examples cited for Sections 15.1.3 and 15.2.1, both units have the same DEHC systems based upon the Ovation platform. For some Chapter 15 events in the UFSAR, there are Unit differences in terms of fuel cycle parameters used as inputs for analyses; however, the analysis results are not used for the D3 analysis. The following sentences will be added to the first paragraph of Section 3: "The UFSAR describes some differences between the two units in fuel cycle parameters used for the Chapter 15 event analyses, The analyses are applicable for both LGS Units 1 and 2 because there are no differences between the two units in terms of functions, sequences of events, control, and protection actions."	OPEN		
17	SNSB		**Coping Analysis:** The D3 CCF coping analysis states [[The Limerick Plant Reference Simulator was used as a tool to guide the analysis for necessary manual operator actions and estimates for time available for these actions. The Limerick operations staffs were used to help assess the acceptability of the coping actions and the timing required for adequate results.]] In the descriptions of all the Chapter 15 events, there is only one event where it is stated that [[Performance of this scenario at the plant simulator demonstrated that the operator actions are appropriate and adequate to maintain the core covered and thus fuel clad temperature below its limit.]] a. [[Were any other events run in the simulator? If so, which ones? b. What documentation is available to describe the simulator runs? **c.** Is there something equivalent to a calculation file/notebook that would have the details as to what was run and the overall results (i.e., sequence of events, figures of water level and reactor coolant system pressure, etc.)?]]	6/2/2022 **a.** [[The following events were run in the simulator: 1. 15.1.3 Pressure Regulator Failure Open 2. 15.2.3 Turbine Trip without Bypass 3. 15.2.4 MSIV Closure 4. 15.2.5 Loss of Condenser Vacuum 5. 15.2.6 Loss of All Grid Connections 6. 15.2.7 Loss of Feedwater 7. 15.4.5 Recirculation Flow Control Failure –Increasing 8. 15.6.4 Steam Line Break Outside Containment 9. 15.6.5 Main Steam Line Break Inside Containment 10. 15.6.5 Recirculation Line Break Inside Containment (DBA) **11.** [[Spurious actuations of CS, HPCI, LPCI, ADS, Drywell Water Cooling, ARI, SLCS, and RRCS]]]] **b.** [[Limerick maintains the simulator runs and can make them available for audit.]] **c.** [[Yes, simulator runs are available for audit.]]	OPEN		

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18	ELTB		[[[redacted]]]	[[[redacted]]]	OPEN		
19	ELTB		**Spurious Actuation Analysis:** [[[redacted]]] Also, there's a minor typographical error in the introductory paragraph of Section 3.30, which refers to "Section 0" for an analysis of spurious initiation of HPCI. There is no Section 0, but spurious HPCI initiation is discussed in the introduction to Section 5, as well as in Section 5.6.	6/8/22: [[[redacted]]] The typographical (Section 0) will be corrected in Section 3.30 to refer to Section 5.6.	OPEN		

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20	EICB		**Reactor Mode Switch:** The analysis in Section 3.23, "Rod Withdrawal Error," refers to automatic control actions (interlocks and rod block signals) that are initiated by the reactor mode switch. The reactor mode switch provides input to the PPS and therefore functions initiated by this switch appear to rely on PPS functionality. Therefore, such functions cannot be credited as mitigation measures when considering common cause failures of the PPS. If the reactor mode switch initiates functions that operate independently from the PPS, then a diversity analysis must be included to demonstrate that such functions remain operable in the presence of a CCF of the PPS. The topical report does not include such an analysis.	Each Reactor Mode Switch position has multiple contacts wired to different systems and only a few are wired to the PPS. For the Reactor Mode Switch Shutdown position, one set of contacts for the Reactor Mode Switch Shutdown Reactor Scram are wired to the PPS. The contacts for the Rod Block Interlock are not wired into the PPS but rather to the Reactor Manual Control System for interlocks. Thus, the rod block functions using separate Reactor Mode Switch contacts in the Reactor Manual Control System remain operable in the presence of a CCF of the PPS.	OPEN		
21	APLC		**Risk Achievement Worth (RAW) Methodology:** Section 4, "BTP 7-19 Position 4 Displays and Controls," of the D3 Analysis examines the five critical safety functions from SECY-93-0087 and defines the diverse controls to achieve each critical safety function and displays to monitor the performance of these functions from the control room. The NRC staff notes that the Limerick probabilistic risk assessment (PRA) results are only referenced in the analysis of the containment isolation critical safety function. It is unclear to the NRC staff why PRA results are included only for this analysis.	6/26/22: The required diverse controls for the other critical safety functions have a manageable set of required controls. In the case of Containment Isolation, there are many isolation functions and not all are risk significant. Therefore, a risk-informed approach was used for Containment Isolation. This is the intent for the introductory paragraph in Section 4.4 that states, "There are numerous conditions that isolate the reactor and containment penetrations. UFSAR Table 6.2 17 provides a list of the isolation valves, automatic isolation conditions, valve tag numbers, etc. The NSSSS automatically closes specific isolation valves upon specific conditions in the reactor or containment." For diverse manual controls required by BTP 7-19 Position 4 for Containment Isolation, the practical approach was to use a risk-informed process for these Position 4 controls.	OPEN		
22	APLC		**RAW Methodology:** The D3 Analysis does not specify what the phrase "RAW of ≥ approximately 2" means or how the RAW values are calculated. The D3 Analysis does not discuss why other risk importance measures (e.g., Risk Reduction Worth, Birnbaum, or Fussell-Vesely) were not considered. The D3 Analysis does not discuss any modifications that were made to the PRA model in order to obtain these RAW values or any peer reviews performed for modifications to the PRA model.	6/17/22: The PCIV risk importance results used are LERF 'RAW' relative risk ranking of both implicitly and explicitly modeled PCIVs within the Limerick internal events PRA models. CDF 'RAW' results are also reviewed, however LERF 'RAW' provided a more expansive listing of risk important PCIVs. The 'RAW' is calculated as the risk increase ratio shown below: $PCIV_i$ D3 RAW = [Total LERF w/PCIVi "Failed Open" with Prob=1] / [base Total LERF] The above PCIV RAW risk importance value determination is different from component importance measure estimates typically obtained directly from base PRA quantification results. FV and other importance measures were not reviewed based on the limitations of the modeling of multiple	OPEN		

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				PCIVs within PRA initiating Events, however it is not anticipated that the FV importance measure would generate a list of risk important PCIVs more expansive than those generated based on RAW importance measures. "RAW of ≥ approximately 2" means that at a minimum, all identified valves with a RAW ≥ 2 are included in section 4.4.1. Additional Valves are included in the section 4.4.1 list to maintain uniformity between system trains that may have unequal importance measures. The Limerick PRA models of record exercised were not modified to perform this evaluation.			
23	APLC		**RAW Methodology:** Section 4.4.1 of the D3 Analysis states that "for those containment penetrations not screened by the PRA, screening is applied to any containment isolation valves determined to be risk insignificant based on the pipe size." Slide 21 for the presubmittal meeting states "For penetrations not screened, additional screening is applied to containment isolation valves determined to be risk significant based on pipe diameter, where small leak failures of containment were risk insignificant." The NRC staff is unable to reconcile the two sentences because one uses "risk insignificant" and the other uses "risk significant" for the size based screening. In addition, the NRC staff is unclear about the purpose and mechanics of this screening.	6/17/22: Both statements identified include general PRA screening criteria used in PRA model development. These statements for size based screening are redundant to those already used in PRA model development. No new or additional size-based screening was performed in the selection of risk important penetrations for this evaluation. Risk significance used in the D3 is based on risk importance measures only. Those PCIVs found to be risk-insignificant based on RAW are further screened, as all PRA modeled PCIVs are initially evaluated for importance measures.	OPEN		
24	APLC		**RAW Methodology:** The presubmittal meeting slides state that "not all containment penetrations or isolation valves are modeled for isolation function."	In general, pipe diameters < 1 inch for water breaks and < 2 inch for steam breaks are risk insignificant and thus are not considered in the PRA analysis for LERF. References: Burns, E. T., et al., ISLOCA Evaluation Guidelines, NSAC-154, September 1991, and McKenna, T.J., Giitter, J.G., "Source Term Estimation During Severe Nuclear Power Plant Accidents", Nuclear Plant Journal, November-December 1988, pp. 83-98. Note that the use of PRA for the D3 analysis is used only to identify diverse Position 4 controls for the critical safety function of primary containment isolation. The PRA identified all of the large bore piping reactor vessel penetrations: 1. HV-041-*F022*, Inboard MSIVs 1. HV-041-*F028*, Outboard MSIVs 2. HV-049-*F007, Inboard RCIC Steam Line Isolation Valve 3. HV-055-*F002, Inboard HPCI Steam Line Isolation Valve	OPEN		

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				4. HV-051-*F008, Outboard Shutdown Cooling Supply Line Isolation Valve 5. HV-051-*F009, Inboard Shutdown Cooling Supply Line Isolation Valve 6. HV-051-*F015A, Outboard Shutdown Cooling Return Isolation Valve 7. HV-051-*F015B, Outboard Shutdown Cooling Return Isolation Valve 8. HV-044-*F001, Inboard RWCU Supply Line Isolation Valve 9. HV-051-*F017*, Outboard LPCI Discharge Isolation Valve The above list confirms the engineering judgement for isolation to limit reactor coolant loss and offsite release. Furthermore, in consideration of small-bore penetrations, leakages from the primary containment to the secondary containment (Reactor Enclosure) will be mitigated by the diverse isolation of Reactor Enclosure air supply and exhaust valves, that will automatically start the Standby Gas Treatment System to limit offsite release below regulatory requirement.			
25	SNSB		D3 Section 3.2, Event: 15.1.2 Feedwater Control Failure – Maximum Demand (Without Turbine Bypass) a. How low would the water level be expected to drop in the 10 minutes plus time for operator action to restore feedwater? What is the basis for the expected minimum water level? Are there any thermal hydraulic calculations or simulator runs performed for this or similar cases where there is a resulting plot of water level -vs- time when there is no feedwater? b. Section 15.1.2.2.2 of the UFSAR states that the low water level initiation of the reactor core isolation cooling system and the HPCI system are to maintain long term water level control following tripping of feedwater pumps. Section 3.2 of the D3 does not mention this and credits operator action to restore feedwater flow. Are RCIC/HPCI assumed failed due to the PPS CCF as they are in other events? c. Does T-101, "LGS RPV Control emergency operating procedure (EOP)," need to be updated to address CCF of the PPS? Specifically, in this case does it need to be updated to account for manual control of automatic depressurization system (ADS), or does it already address use of the ADS? Do other EOPs need to be updated? d. Confirm that the RRCS high pressure trip setpoint is 1,149 psig (Technical Specification Table 3.3.4.1 2) and that this initiates alternate	6/24/22 a. Assuming no operator action in the first 10 minutes after the high level trip (Level 8) of the reactor feedpump turbines, the minimum water level is approximately -95 inches on the wide range instrument which is 34 inches above Level 1. Simulator runs with plots of water level and other pertinent parameters are available for audit. b. RCIC and HPCI control functions are in PPS. These systems would be assumed to fail to start when RPV water level decreases to Level 2. For this transient, RPV water level increases to high level (Level 8) which trips the feedpump turbines. Operator action is credited for restoring water level using manual Woodward feedpump turbine speed control system. c. The EOPs do not need to be updated. Operator actions	OPEN		

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			rod insertion as LCO 3.3.4.1 only mentions RPT. e. How long does it take for operators to restore feedwater pumps?]]	are based upon symptoms, not event specific. T-101 specifies the use of feedpumps, HPCI, RCIC, condensate pumps, etc., to maintain water level. The use of ~~ADS~~ SRVs is specified in the reactor pressure control portion of the procedure. d. RRCS high pressure trip setpoint is 1,149 psig which will also initiate ARI. e. Simulation of this event indicated that the operator can reset the feedpump turbine trips and restore the feedpumps to inject into the RPV in less than 3 minutes, using the manual Woodward turbine speed control system.]]			
26	SNSB		D3 Section 3.3, Event: 15.1.2 Feedwater Control Failure – Maximum Demand (With Bypass) [[Section 3.2.4 (without bypass) states operators restore feedwater using "manual Woodward speed control" of feedwater system pumps. However, Section 3.3.4 (with bypass) states "using manual control of the feedwater system." Is there an actual difference that the operators would take, or is it just the wording that is different?]]	There is no actual difference that the operators would take in both instances. Section 3.3.4 will be revised to use "manual Woodward speed control" of feedwater system pumps."	OPEN		
27	SNSB		D3 Section 3.4, Event: 15.1.3 Pressure Regulator Failure-Open a. [[Section 15.1.3 of the LGS UFSAR provides some differences between Unit 1 and Unit 2 operation for this event. Please address the differences (perhaps generically) and confirm that the results presented are valid for both Unit 1 and Unit 2. b. How low would the water level be expected to drop before operator actions get the system pressure below the shutoff head of the condensate pumps? What is the basis for the expected minimum water level? Assuming the operators start depressurizing at 10 minutes, how long does it take to get the pressure below the shutoff head of the condensate pumps? Are there any TH calculations or simulator runs performed for this or similar cases where there is a plot of water level -vs- time when there is no feedwater? c. If level gets too low (< 186 in), would operators do an emergency depressurization with ADS to get pressure below shutoff head of pump?]]	a. There are no differences between LGS Units 1&2 in terms of system response, functions, and accident sequences and thus the analyses are applicable for both units. In the examples cited for Sections 15.1.3 and 15.2.1, both units have the same DEHC systems based upon the Ovation platform. For some Chapter 15 events in the UFSAR, there are Unit differences in terms of fuel cycle parameters used as inputs for analyses; however, the analysis results are not used for the D3 analysis. The following sentences are added to the first paragraph of Section 3: "The UFSAR describes some differences between the two units in fuel cycle parameters used for the Chapter 15 event analyses. The analyses are applicable for both LGS Units 1 and 2 because there are no differences between the two units in terms of functions, sequences of events, control, and	OPEN		

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				protection actions." 6/24/22 b. The DPS automatically closed the MSIVs upon low pressure conditions. Assuming no operator action in the first 10 minutes after initiation of this event, the RPV water level decreased to approximately -130 inches. After 10 minutes, the operator started depressurizing the vessel using the 3 SRVs which took approximately 3 minutes to the condensate pump shutoff head of 680 psig. The condensate pumps were used to reflood the vessel. The minimum water level reached is approximately -190 inches on the fuel zone instrument which is below TAF (-161 inches). Although water level dropped below TAF, the PCT remained below the initial PCT value prior to the event due to steam cooling. Simulator runs with plots of water level, pressure, and PCT and other pertinent parameters are available for audit. c. Operator would not perform an emergency depressurization with the ADS SRVs for this event if operator action is assumed prior to 10 minutes from initiation of the event. Upon entry to the EOPs for RPV pressure control, the operator would use the ADS SRVs to stabilize pressure below 1096 psig, and to depressurize the RPV at a cooldown rate below 100°F/hr. Once RPV pressure is below the condensate pump shutoff head, the condensate pumps can inject to restore water level. Water level was recovered to above TAF in approximately 1 minutes after the condensate pumps started to inject.			

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28	SNSB		D3 Section 3.5, Event: 15.1.4 Inadvertent Main Steam Relief Valve Opening a. {{ Are the turbine bypass valves/controls independent of the PPS (they are not listed in Section 3.5.5)? b. Section 3.5.2 states that no safety control action is needed, however, the operators will need to initiate a manual scram when the suppression pool temperature increases above 110°F. **c.** One of the operator actions in Section 3.5.4 is "Emergency RPV Depressurization is Required." Is this done with the ADS SRVs? If so, they aren't listed as available in Section 3.5.5. }}	a. {{ The turbine bypass valves/controls are part of the DEHC (Digital Electro-Hydraulic Control system) using the Ovation platform. DEHC is listed in Section 3.5.5. DEHC performs the reactor pressure and turbine control functions described in UFSAR Section 7.7.1.5 Pressure regulator and turbine-generator system. b. Section 3.5.2 lists the automatic control actions. Because the operator will need to initiate a manual scram, it is not listed in Section 3.5.2, but is listed in Section 3.5.4(1)(a). ~~b.~~c. 6/17/22: Simulation indicated that the HCTL is not exceeded and thus emergency depressurization is not required. Step 3.5.4(1)b will be deleted.}}	OPEN		
29	SNSB		D3 Section 3.6, Event: 15.1.6 Inadvertent RHR Shutdown Cooling Operation {{ Since this event takes place over an extended time, when does the 10 minute clock start? Is it possible there would be no notification that anything is wrong until 10 minutes? Should the clock start once it is noticed that something is wrong? In most events, things happen within seconds so the operators are aware almost immediately, so the 10 minute clock can start at event initiation. In the UFSAR analysis, the flux scram is not predicted to occur within the 10 minutes. }}	{{ Based upon the sequence of events in Section 3.6.1 for this event, time zero is considered to be the start of the event. The slow reactor power increase due to moderator temperature decrease would result in a slow increase in reactor power as monitored by the IRMs. The operator would be monitoring shutdown cooling operation and reactor power, and thus increases in temperature or reactor power will alert the operator to take the appropriate action such as insertion of control rods. If the operator does not take action, then an automatic reactor trip would occur on IRM Upscale, well beyond 10 minutes after event initiation. 6/17/22: The IRM upscale alarm will be added to Section 3.6.5.}}	OPEN		
30	SNSB		D3 Section 3.7, Event: 15.2.1 Pressure Regulator Failure – Closed {{ Pressure control system is listed in Section 3.7.2, however, it is not included in Section 3.7.5. Is the pressure control system diverse from PPS? }}	{{ The pressure control system is DEHC and is a diverse system from PPS. See response to Question 28(a) above. Section 3.7.2 will be revised to use the term "DEHC" for the pressure control system and listed in Section 3.7.5. }}	OPEN		

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31	SNSB		D3 Section 3.8, Event: 15.2.2 Generator Load Rejection with Bypass Failure UFSAR Section 15.2.2.3.2 states "Actual closure of main steam isolation valves (MSIVs) as caused by low water level trip (Level 1), and actual flow from initiation of RCIC and HPCI core cooling system functions do not occur during the duration of the simulation. If these events occur, they will follow sometime after the primary concerns of fuel margin and overpressure effects have passed and are expected to result in effects less severe than those already experienced by the reactor system." Can the digital feedwater and level control system maintain water level long term? What is the source of water into the RPV if water level does decrease and MSIVs close causing loss of the normal feedwater pumps? Are RCIC and HPCI assumed failed given the PPS CCF?	6/24/22: For this event, the turbine bypass valves are assumed to fail to open. The reactor will remain at high pressure after the scram due to reactor decay heat. Thus, the DFWLCS can continue to use the feedpump turbines to make up for inventory loss through operation of the SRVs, and maintain RPV water level. One of the EOP entry conditions is RPV water level low (Level 3). Upon EOP entry, the operator is instructed to depressurize the reactor at a rate less than 100 deg. F per hour, and maintain water level using available injection sources. When RPV pressure decreases below the condensate pump shut off head (approximately 680 psig), the condensate pumps can inject water to maintain RPV water level. RCIC and HPCI are assumed to have failed due to the PPS CCF. However, a diverse Core Spray loop taking suction from the suppression pool will be available to inject water into the RPV with the reactor depressurized below the shutoff head of the Core Spray pump (approximately 335 psig). The condenser hotwell capacity is at least 3 minutes of full-power operation, assuming no makeup to the hotwell, equivalent to approximately 100 minutes of operation at a decay heat level of 3%. The time to depressurize the reactor to the Core Spray pump shutoff head is estimated to be approximately 77 minutes, assuming a cooldown rate of 90 deg. F per hour. Furthermore, the condensate storage tank can make up water (approximately 60,000 gallons) to the condenser hotwell to maintain hotwell water level (UFSAR Section 10.4.7.5). Condenser hotwell water level is a Regulator Guide 1.97 parameter (UFSAR Section 7.5.2.5.1.1.2.4.9) monitored at the main control room. Thus, the condenser hotwell, along with the Core Spray system taking suction from the suppression pool, has sufficient capacity to enable long term availability of water to maintain RPV water level after a reactor scram with the postulated failure of bypassing steam to the main condenser.	OPEN		

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32	SNSB		D3 Section 3.9, Event: 15.2.2 Generator Load Rejection with Bypass Section 3.9.2, Automatic Control Actions in the presence of postulated PPS CCF, diverse from PPS, shows the pressure control system modulates turbine bypass valves to control pressure. Section 3.9.4, Operator Actions per RPV Control / Primary Containment Control EOP with postulated PPS CCF, states to stabilize RPV pressure at a pressure below 1,096 psig using the turbine bypass valves. Is this an automatic action, or operator controlled action?	This can be an automatic or manual control action. DEHC automatically modulates the turbine bypass valves after a load rejection to control pressure to an operator entered setpoint value at the DEHC HMI. The operator has the option to use the turbine bypass valves to control pressure manually.	OPEN		
33	SNSB		D3 Section 3.12, Event: 15.2.4 MSIV Closure How low would the water level be expected to drop before operator actions get the system pressure below the shutoff head of the condensate pumps? What is the basis for the expected minimum water level? Assuming the operators start depressurizing at 10 minutes, how long does it take to get the pressure below the shutoff head of the condensate pumps? Are there any TH calculations or simulator runs performed for this or similar cases where there is a plot of water level -vs- time when there is no feedwater?	6/21/22: Assuming no operator action in the first 10 minutes initiation of this event, it would take approximately 3 minutes to depressurize the reactor using 3 SRVs to the condensate pump shutoff head of 680 psig. The minimum water level reached is approximately -65 inches on the wide range instrument which is above Level 1 (-129 inches) by approximately 64 inches. Water level did not drop below TAF, thus PCT remained below the initial PCT value prior to the event. Simulator runs with plots of water level, pressure, and PCT and other pertinent parameters are available for audit.	OPEN		
34	SNSB		D3 Section 3.13, Event: 15.2.5 Loss of Condenser Vacuum How low would the water level be expected to drop before operator actions get the system pressure below the shutoff head of the condensate pumps? What is the basis for the expected minimum water level? Assuming the operators start depressurizing at 10 minutes, how long does it take to get the pressure below the shutoff head of the condensate pumps? Are there any TH calculations or simulator runs performed for this or similar cases where there is a plot of water level -vs- time when there is no feedwater?	6/21/22: Assuming no operator action in the first 10 minutes initiation of this event, it would take approximately 3 minutes to depressurize the reactor using 3 SRVs to the condensate pump shutoff head of 680 psig. The minimum water level reached is approximately -95 inches on the wide range instrument which is above Level 1 (-129 inches) by approximately 34 inches. Water level did not drop below TAF, thus the clad temperature increased above the initial clad temperature prior to the event by approximately 20 deg. F, due to the main turbine trip with no bypass operation because bypass is inhibited from opening on low vacuum. Simulator runs with plots of water level, pressure, and PCT and other pertinent parameters are available for audit.	OPEN		

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35	SNSB		D3 Section 3.14, Event: 15.2.6 Loss of All Grid Connections [[How low would the water level be expected to drop before operator actions get the system pressure below the shutoff head of the core spray pumps? What is the basis for the expected minimum water level? Assuming the operators start depressurizing at 10 minutes, how long does it take to get the pressure below the shutoff head of the core spray pumps? Are there any TH calculations or simulator runs performed for this or similar cases where there is a plot of water level -vs- time when there is no feedwater? Why are the core spray pumps used for this event while condensate pumps are used in other events?]]	6/21/22: Assuming no operator action in the first 10 minutes initiation of this event, it would take approximately 3 minutes to depressurize the reactor using the ADS SRVs to the Core Spray pump shutoff head of 335 psig. The minimum water level reached is approximately -235 inches on the fuel zone instrument which is below TAF (-161 inches) by approximately 74 inches. Water level dropped below TAF, thus the clad temperature increased above the initial clad temperature prior to the event by approximately 20 deg. F. This was due to the main turbine trip with no bypass operation because bypass is inhibited from opening on low vacuum. During the time that the core is uncovered, PCT is maintained by steam cooling until reinjection. Simulator runs with plots of water level, pressure, and PCT and other pertinent parameters are available for audit.	OPEN		
36	SNSB		D3 Section 3.15, Event: 15.2.7 Loss of Feedwater [[How low would the water level be expected to drop before operator actions get the system pressure below the shutoff head of the core spray pumps? What is the basis for the expected minimum water level? Assuming the operators start depressurizing at 10 minutes, how long does it take to get the pressure below the shutoff head of the core spray pumps? Is there any information from the simulator run performed for this case that shows a plot of water level -vs- time?]]	6/21/22: Assuming no operator action in the first 10 minutes initiation of this event, it would take approximately 3 minutes to depressurize the reactor using the ADS SRVs to the Core Spray pump shutoff head of 335 psig. The minimum water level reached is approximately -245 inches on the fuel zone instrument which is below TAF (-161 inches) by approximately 84 inches. Although water level dropped below TAF, the PCT remained below the initial PCT value prior to the event. During the time that the core was uncovered, PCT is maintained by steam cooling until reinjection. Simulator runs with plots of water level, pressure, and PCT and other pertinent parameters are available for audit.	OPEN		
37	SNSB		D3 Section 3.16, Event: 15.2.9 Loss of Shutdown Cooling a. [[Section 3.16.4 of the D3 analysis states "Upon recognition of loss of shutdown cooling:" What procedure are the operator actions based on? b. Is suppression pool cooling initiation diverse from PPS?]]	a. [[The operator actions are based upon abnormal operating procedure ON-121 Loss of Shutdown Cooling, to establish alternate shutdown cooling. This procedure will be referenced in 3.16.4 and will be made available for audit. **b.** Suppression pool cooling initiation is diverse from PPS. Step 3 in Section 3.16.4 to initiate suppression pool cooling was inadvertently deleted from the report and will be restored in the next revision of the document.]]	OPEN		

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38	SNSB	D3 Section 3	General Questions a. [[Is the Level 8 Main Turbine Trip and Feedwater Pump Turbines Trip independent of PPS~~?~~]] **b.** [[Is there a diverse action for closing the MSIVs (as described in 3.1.32) available during other sequences? What are the actuating conditions?]] **c.** [[Are the Manual SCRAM push buttons and/or Mode Switch separate and diverse from PPS? Or otherwise excluded by PPS CCF?]] d. [[Is the Core Spray initiation push button fully separate and diverse from PPS? Or otherwise excluded by PPS CCF?]]	a. The Level 8 trip is independent of PPS but it is using an AC70 PLC which is not considered sufficiently diverse to be credited in the analysis. WEC presents revised analyses at the 6/16/22 meeting to address this. b. Automatic closure of the MSIVs ~~are~~is summarized in Table 3-1. Manual closure of the MSIVs will also be available at the DPS HMI. c. The Manual SCRAM pushbuttons are separate and diverse from PPS software, and are hardwired directly to the RPS Termination Units in PPS bypassing all software. This is described in Section 2.2.3 PPS Diversity. d. The A Core Spray loop is incorporated in the separate and diverse DPS, including system initiation. The analyses in this report identified the need for a diverse CS initiation logic system to mitigate specific events in Chapter 15 of the UFSAR.	OPEN		
39	SNSB	D3 Section 3.19	Trip of Both Recirculation Pumps a. [[What are the inventory control and dose consequences of delaying the SCRAM from Level 8 at 5s to Level 2 at 43s? Will the MSSVs pass additional steam or water inventory?]] b. [[Is HPCI or RCIC manual actuation available as a mitigating strategy to the operators?]] [[Why is >10 minutes acceptable response timeline for operator action with no feed or HPCI/RCIC?]]	a. There will be no inventory control and dose consequences of delaying the SCRAM until RRCS actuates the ARI valves, because water level is automatically controlled by the DFWLCS and core is covered at the normal water level. b. HPCI and RCIC functions are incorporated in PPS and assumed to not respond due to the postulated PPS CCF. c. The DFWLCS maintains water level in accordance with EOPs. Feed from HPCI/RCIC is not required. 6/22/22: Although procedure OT-112, "Unexpected/Unexplained Change in Core Flow", requires an immediate manual scram, <5 minutes response time will be specified.	OPEN		
40	SNSB	D3 Section 3.26	Abnormal Idle Recirculation Pump Start **a.** [[What actually SCRAMs the Reactor in the absence of APRM upscale? Hi pressure? Low Level? Operator Action? (Conclusion says operator manual scram) When does this occur sequence timeline?]] [[Why is >10min for operator action an acceptable criteria when the previous analysis assumed PPS action in 10 seconds?]]	6/29/22: Simulation of this event indicates that reactor power rises to approximately 72% from a nominal power of 55%, and stabilizes at approximately 66% power, when the idle recirculation pump is started, well below the high flux or high thermal power reactor scram setpoints. Note that the ASD starts with the following speed change rates to the minimum speed of 466 rpm: 0-333 rpm at 84 rpm/second, >333 rpm at 42 rpm/second. This "soft start" algorithm minimizes the reactor power peak upon a pump start. The DFWLCS and DEHC maintain water level and reactor pressure, respectively, and thus no operator action is necessary. The analysis of this event will be revised accordingly.	OPEN		

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41	SNSB	D3 Section 3.27	Recirculation Flow Control Failure with Increasing Flow **a.** [[What actually SCRAMs the Reactor in the absence of APRM upscale? Hi pressure? Low Level? Operator Action? (Conclusion says operator manual scram) When does this occur sequence timeline?]] [[Why is >10min for operator action an acceptable criteria when the previous analysis assumed PPS action in <2 seconds?]]	This event was simulated, assuming no reactor scram on high neutron flux. Reactor pressure was automatically controlled by DEHC, by fully opening the turbine control valves and modulation of the main turbine bypass valves. Reactor water level was adequately controlled automatically by the DFWLCS. Neutron flux increased above the normal high flux trip setpoint but returned to a nearly stable value below the high flux trip setpoint. No scram occurred due to high pressure or low water level. Upon entry into the EOPs, the operator would perform a manual scram at greater than 10 minutes	CONFIRM		
42	SNSB	D3 Section 3.29	Control Rod Drop Accident a. [[Section 3.29.4 specifies operator manual scram per OT-117 RPS Failures. When does this occur sequence timeline?]] [[Why is >10min for operator action an acceptable criteria when the previous analysis assumed PPS action in <2 seconds?]]	a. Upon receipt of an IRM Upscale alarm immediately after the postulated control rod drop, and recognition that the reactor did not scram, the operator will take immediate action to perform a manual scram per OT-117 RPS Failures. b. A greater than 10 minute operator response is acceptable because the dropping of a rod results in a high local reactivity in a small region of the core, but the average reactor power as measured by the APRM increases slowly beyond 10 minutes. The slow core average power increase after the control rod drop poses no threat to the reactor system.	OPEN		
43	SNSB	D3 Section 3.32	Main Steam Pipe Break Outside Containment **a.** [[Manual Actions are being substituted depressurization and aligning ECCS what is the impact of the timing delay on core uncovery, minimum level, and reflood? Is the time assumed >10min?]] b. [[A diverse automatic MSIV closure is mentioned in the conclusion, what instrumentation condition causes this? When does this occur in the sequence?]] **c.** [[Does crediting CREFAS and SGTS for this sequence affect the systems overall risk importance? Safety Related and in TS? What is are the systems history of reliability?]]	a. If the operator takes no action for the first 10 minutes of the event, the minimum water decreases to approximately -100 inches (wide range), with reactor pressure control by automatic cycling of the SRVs. Following instructions in the EOPs, an emergency depressurization by opening of all ADS SRVs at 10 minutes resulted in water level dropping below TAF. However, use of the condensate pumps reflooded the core to normal water level in less than 5 minutes after the emergency depressurization. The PCT remained below its acceptable value through the event. b. As indicated in Section 3.32.5(14), automatic MSIV closure occurs due to sensed high main steam line flow. This automatic MSIV closure is performed in DPS. This automatic MSIV closure would occur approximately 1 second after initiation of this event as indicated in 3.32.1 Sequence of Events. c. 6/28/22: Neither CREFAS ~~or~~nor SGTS are credited in the accident analysis, as stated in UFSAR section 15.6.4.5 and are not listed as credited systems for coping in this analysis.	OPEN		

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Limerick D3 Analysis Evaluation Open Item Summary Table							
Item No.	**Source Branch**	**Location in Application**	**Issue Description** *(From NRC Staff)*	**Licensee Response**	**Status** *(From NRC Staff)*	**Audit, RAI, or RCI Number** *(From NRC Staff)*	**Licensee Supplement** *(if any)*
44	SNSB	D3 Section 3.33	D3 Section 3.33 LOCA Inside Containment (Recirc Line) **a.** {{Manual Actions are being substituted depressurization and aligning ECCS what is the impact of the timing delay on core uncovery, minimum level, and reflood (in 130s)? Is the time assumed >10min?}} **b.** {{Is the push button Core Spray initiation fully separate and diverse from PPS? If not how much time will full manual alignment require?}}	a. For this event, no manual operator action is needed to depressurize the reactor because the reactor will depressurize through the recirculation line break. Blowdown of the reactor will take less than one minute. The diverse CS system logic at the DPS will automatically initiate upon high drywell pressure and reactor low pressure, and therefore, no manual ECCS alignment would be necessary to reflood the core to 2/3 core height. With one loop of core spray operating, spray flow and 2/3 core height submergence in approximately 13-14 minutes limited the peak PCT to approximately 1550 deg. F. b. The diverse CS system logic is implemented in DPS for automatic initiation and no manual alignment would be needed. Manual system initiation is also available at the DPS HMI. DPS is completely separate and diverse from PPS as discussed in Section 6 of the D3 report.	OPEN		

Limerick D3 Analysis Evaluation Open Item Summary Table

Item No.	Source Branch	Location in Application	Issue Description *(From NRC Staff)*	Licensee Response	Status *(From NRC Staff)*	Audit, RAI, or RCI Number *(From NRC Staff)*	Licensee Supplement *(if any)*
45	SNSB	D3 Section 3.34	LOCA Inside Containment (Main Steam) **a.** [[Manual Actions are being substituted depressurization and aligning ECCS what is the impact of the timing delay on core uncovery, minimum level, and reflood Is the time assumed >10min?]] **b.** [[Does crediting SLCS and SGTS for this sequence affect the systems overall risk importance? Safety Related and in TS What is are the systems history of reliability?]]	a. For this event, no manual operator action is needed to depressurize the RPV because depressurization is achieved through the main steam line break inside containment. With automatic CS initiation and condensate pump injection, the RPV is flooded to the level of the assumed steam line break. The MSIVs automatically closed on low main steam line pressure. The core was never uncovered for this event and thus PCT remains below its initial value. Reactor power was reduced by recirculation pump trip, depressurization, and decrease in recirculation flow and stabilized at approximately 25% beyond 10 minutes from event initiation. RRCS initiating conditions (low water level, high pressure) were not reached and thus operator action to scram the reactor is necessary upon entry into the EOP. Estimated Time Available for Operator Actions is revised to be < 5 minutes from event initiation. Although this time can be chosen to be longer than 10 minutes, 5 minutes is conservatively chosen for the operator to perform a manual scram to terminate the energy discharge to the containment. This time will result in ample margins to the containment design limits, and can be validated to be achievable. b. 6/28/22: The SLCS initiation is performed at RRCS. The LTR provides the safety case for the Ovation-based RRCS being of sufficient quality for ATWS functions. SLCS is a safety related system that has tech spec surveillance requirements to ensure reliable operation. That is unchanged in regards to the DMP. ~~3.34 is revised to show that SGTS is not credited in the radiological release analysis and therefore operation of SGTS and RERS are not specified~~ 3.34.5 does not indicate SGTS and RERS as diverse backup systems because the radiological release analysis did not credit the operation of these systems. However, SLCS, SGTS and RERS are safety related systems and have tech spec surveillance requirements for ensure reliable operation, if needed.	OPEN		

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Limerick D3 Analysis Evaluation Open Item Summary Table							
Item No.	**Source Branch**	**Location in Application**	**Issue Description** *(From NRC Staff)*	**Licensee Response**	**Status** *(From NRC Staff)*	**Audit, RAI, or RCI Number** *(From NRC Staff)*	**Licensee Supplement** *(if any)*
46	SNSB	D3 Section 3.35	Feedwater Line Break Outside Containment {{Why does section 3.35.4 direct the operators to depressurize and initiate CS, instead of HPCI and RCIC?}}	HPCI and RCIC functions are implemented in PPS. With the assumption of a CCF in PPS, it was assumed that both HPCI and RCIC would not respond. Additionally, condensate is not available due to the feedwater line break. With the reactor initially at high pressure after initiation of this event, it is necessary to depressurize the reactor to below the shutoff head of the CS pumps (approximately 335 psig) and allow the diverse CS system control at DPS to inject water into the reactor.	OPEN		
47	EICB	D3 Sections 3.46, 4.7, and 6.5	**Shared from PPS:**	In Table 3-2, of the D3 Analysis certain sensors are indicated as "To be shared from PPS". These sensors are shared from the PPS, by terminating the sensor signal in the PPS cabinet and cabling the signal to both PPS and to analog input modules at the DCS Remote Node Interface (RNI) also located in PPS cabinets. The output signals from these RNI analog input modules are transmitted to DPS via fiberoptic cables for electrical isolation and separation. The signals are from analog sensors, so a CCF of the sensors is not considered. The sensor signals may be displayed on the DPS HMI screens, and thus their entries in the Instrument Tag # and Panel # Location columns are blanked. Table 3-3 summarizes the sensors required for each automatic function specified in Table 3-1. The PPS Licensing Technical Report provides the detailed description of these shared inputs. Power for the shared sensors is provided by diverse power supplies in the PPS cabinet to avoid a CCF.	OPEN		
48	EICB	Section 5.9.4	[[[redacted] [redacted] [redacted]]]	6/23/22: [[[redacted]]]	OPEN		

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Item No.	**Source Branch**	**Location in Application**	**Issue Description** *(From NRC Staff)*	**Licensee Response**	**Status** *(From NRC Staff)*	**Audit, RAI, or RCI Number** *(From NRC Staff)*	**Licensee Supplement** *(if any)*
49	EICB	Section 5.4	[[[redacted]]]	[[[redacted] [redacted]]]	OPEN		
50	EICB	Section 5.10	[[[redacted]]]	[[[redacted]]]	OPEN		

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NRC Branches: EICB – Electrical and Instrumentation Control Branch
ELTB – Long Term Operations and Modernization Branch
SNSB – Nuclear Systems Performance Branch
IOLB – Operator Licensing and Human Factors Branch
APLC – PRA Licensing Branch
LPL – Plant Licensing Branch

Audit List Item	Source Branch	Requested Information to be Available on Portal for Audit	Reference to Open Item #	Available (Y/N)	Portal Location
1	EICB	Design information on Reactor Mode Switch	20	Y	
2	EICB	Design information on configuration of CIM to support diversity from PPS	3, 6	Y	WEC
3	EICB	Design information on HARP module to support diversity from PPS evaluation	7	Y	WEC
4	EICB	DPS Function Allocation / Design information to support diversity from PPS evaluation	8	Y	WEC
5	EICB	Information on credited control and indications that are not DPS to support diversity between systems performing functions and PPS	14	N	
6	SNSB	Documentation to describe simulator runs	17	N	
7	SNSB	Reference 8, NEDO-24011-A-16-US, Revision 16, “Supplement to General Electric Standard Application for Reactor Fuel, GESTAR II, Base Document NEDO-24011-P-A,” General Electric July 2006	-----	N	
8	SNSB	Reference 9, OT-101, Revisions 39, “LGS High Drywell Pressure Operating Procedure” Attachment 3 “Loss of Drywell Cooling”	-----	Y	CEG
9	SNSB	Reference 11, OT-110, Revision 24, “LGS Reactor High Level Operating Procedure”	-----	Y	CEG
10	SNSB	Reference 12, OT-117, Revision 12, “LGS RPS Failures Operating Procedure”	-----	Y	CEG
11	SNSB	Reference 13, ON-113, Revision 28, “LGS Loss of RECW Operation Procedure”	-----	Y	CEG
12	SNSB	Reference 14, T-101, Revision 28, “LGS RPV Control EOP”	-----	Y	CEG
13	SNSB	Reference 15, T-102, Revision 28, “LGS Primary Containment Control EOP”	-----	Y	CEG
14	SNSB	Reference 16, T-103, Revision 25, “LGS Secondary Containment Control EOP”	-----	Y	CEG
15	SNSB	Reference 17, T-121, Revision 0, “LGS RPV Control – OPCON 4 EOP”	-----	Y	CEG
16	SNSB	Reference 18, T-131, Revision 0, “LGS Decay Heat Control – OPCON 5 EOP”	-----	Y	CEG
17	SNSB	Reference 36, NEDO-2422, “Assessment of BWR [Boiling Water Reactor] Mitigation of anticipated transient without scram, Volume II (NUREG 0460 Alternate No. 3,” Nonproprietary version, General Electric, February 1981	-----	N	
18	SNSB	Reference 38, SE-10, Revision 65, LGS Special Event Procedure	-----	Y	CEG
19	SNSB	Reference 39, LM-0642, “Suppression Pool pH Calculation for Alternate Source Terms,” Exelon	-----	Y	CEG
20	SNSB	Reference 42, TP18-1-008, Revision 4, “BWROG Emergency Procedure and Severe Accident Guidelines Appendix B Technical Basis, Volume I: Introduction and References,” June 1, 2018	-----	N	
21	SNSB	NEDC-30936P-A (Referenced on page 3-26 of D3)	-----	Y	CEG
22	SNSB	NEDO-24708A (Referenced on page 3-26 of D3)	-----	Y	CEG