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VICE PRESIDENT
NUCLEAR

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Mr. B. J. Youngblood, Chief
Licensing Branch No. 1
Division of Licensing
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Perry Nuclear Power Plant
Docket Nos. 50-440; 50-441
SER Open Issues Nos. 13 and 14
Instrumentation and Control Systems

Dear Mr. Youngblood:

This letter forwards our responses to NRC Questions 420.03, 420.05, and 420.06. These questions are also identified as SER Open Item No. 13, "Loss of Non-Class 1E Instrumentation and Control Power System Bus During Operation" and Open Item No. 14, "Control System Failure".

We believe this information is sufficient to allow for your review and resolution of these issues. If you have any questions, please contact me.

Very truly yours,

Murray R. Edelman

Murray R. Edelman
Vice President
Nuclear Group

MRE:kh

cc: Jay Silberg, Esq.
John Stefano
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Enclosures

B001

QUESTION

420.03

ITEM 1:

Review the Class 1E and Non-Class 1E buses supplying power to safety and nonsafety-related instrumentation and control systems which could affect the ability to achieve a cold shutdown condition using existing procedures or procedures developed under Item 2 below. For each bus:

a) Identify and review the alarm and/or indication provided in the control room to alert the operator to the loss of power to the bus.

b) Identify the instrument and control system loads connected to the bus and evaluate the effects of loss of power to these loads including the ability to achieve a cold shutdown condition.

c) Describe any proposed design modifications resulting from these reviews and evaluations, and your proposed schedule for implementing those modifications.

RESPONSE

CEI has reviewed the plant design and has concluded that the failure of a single nonsafety class instrumentation or control power bus does not adversely affect the ability to achieve a cold shutdown condition.

Discussion of Class 1E Buses

The PNPP electrical system used for achieving safe shutdown is a Class 1E redundant system. Each redundant division contains its own power sources and separate controls and instrumentation. The trains are both electrically independent, and physically separated. Additionally, the PNPP is designed to Regulatory Guide 1.75, therefore, a failure of a non-Class 1E electrical component or system cannot adversely impact the operation of the Class 1E electrical control system. Instrumentation and controls required for operation of the safety-related fluid systems are Class 1E, therefore, the only possible means of interaction between safety and nonsafety systems would be through the fluid process system. This, in fact, is the mechanism through which the examples cited in I&E Bulletin 79-27 and I&E Notice 79-22 occurred. It is also the mechanism which was manifest in the February 20, 1980 occurrence at Crystal River and in several other occurrences which may be categorized as systems interaction events.

QUESTION

420.03

ITEM 2:

Prepare emergency procedures or review existing ones that will be used by control room operators, including procedures required to achieve a cold shutdown condition, upon loss of power to each Class 1E and non-1E bus supplying power to safety and nonsafety-related instrument and control systems. The emergency procedures should include:

1) The diagnostics/alarms/indicators/symptoms resulting from the review and evaluation conducted per Item 1 above.

2) The use of alternate indication and/or control circuits which may be powered from other non-Class 1E Class 1E instrumentation and control buses.

3) Methods for restoring power to the bus.

Describe any proposed design modification or administrative controls to be implemented resulting from these procedures, and your proposed schedule for implementing the changes.

RESPONSE

The failure of a single Class 1E and non-Class 1E instrumentation or control power system bus will not adversely affect the ability to achieve a cold shutdown condition. However, CEI realizes that such failures could be undesirable when compounded by operator actions. Therefore, when emergency procedures are prepared, they will include the appropriate operator actions to account for this situation. Perry Plant personnel are in the process of preparing these procedures and are expected to be complete six months prior to fuel load.

QUESTION
420.03

ITEM 3:

Re-review IE Circular 79-02, Failure of 120 Volt Vital AC Power Supplies, dated January 11, 1979, to include both Class 1E and non-Class 1E safety-related power supply inverters. Based on a review of operating experience and your re-review of 1E Circular No. 79-02, describe any proposed design modifications or administrative controls to be implemented as a result of the re-review.

RESPONSE

Perry Plant does not employ the SCI inverters outlined in this bulletin. We use the Cyberex, Inc. inverter and static switch arrangement in a non-Class 1E application. PNPP does employ small inverters in a Class 1E application, however, these inverters do not use the static switch arrangement. Preoperational testing on the non-Class 1E inverters will be performed to ensure maximum possible availability during transient conditions.

QUESTION

420.05

If control systems are exposed to the environment resulting from the rupture of reactor coolant lines, steamlines or feedwater lines, the control systems may malfunction in a manner which would cause consequences to be more severe than calculated in safety analyses. This concern was addressed in IE Information Notice 79-22.

Provide the results of an analysis of interactions between nonsafety grade or control equipment to demonstrate they will not cause consequences more severe than those found in safety analyses when subjected to the harsh environment of a high energy line break.

APPROACH

To evaluate this type of interaction CEI has: 1) identified the functions required to prevent transients; 2) identified all the major components of systems which could significantly impact those functions; 3) identified control and instrumentation dependencies which could cause the components or systems to interact resulting in transients not previously analyzed; 4) performed a detailed evaluation to determine whether the dependencies identified in Step 3 could actually cause the adverse states identified in Step 2.

The functions which require control to maintain the reactor in a safe condition are reactivity, reactor vessel level, reactor system pressure, and decay heat removal. In order to identify those systems which impact these functions, CEI developed a control volume of the reactor vessel, showing all the systems which connect to it.

There is a spectrum of possible flow rates to and from the reactor vessel, ranging from a few gpm for the sampling system to tens of thousands of gpm for the feedwater system. The systems with the largest capacity will cause the quickest and greatest impact on the required safety functions. The systems which were evaluated and the ways in which they impact the functions given above are discussed below:

1) The standby liquid control system inserts negative reactivity into the core. The insertion of negative reactivity is always a conservative step in an accident or transient analysis. The SLC system cannot remove water; it is a manually initiated system which is used strictly as a backup to a redundant safety-related system. Its failure or malfunction will have no adverse impact on plant safety.

2) The control rod drive system is required to assure scram capability. The components required to scram the reactor are redundant and Class 1E. The control rod drive pumps are non-1E and pump a relatively small quantity of fluid into the reactor when compared with the feedwater system. Any failure of the non-1E portions will have no adverse impact on plant safety.

3) The Reactor Plant sampling system is a low capacity system which is manually controlled. The system is automatically isolated by redundant Class 1E isolation valves. There are no non-1E failures associated with this system which could have an adverse impact on plant safety.

4) The reactor water cleanup system is a normally operating system with minimal capacity compared to the feedwater system. The system is automatically isolated by redundant Class 1E isolation valves on low reactor water level. The failure or malfunction of non-1E portions of this system will not seriously impact plant safety.

5) The head vent system valves are manually operated. Valve position is not affected by a bus or control system failure.

The conservatisms in the design of the safety-related systems are almost certain to absorb the small system effects from the above systems. The systems remaining to be evaluated are:

1) Recirculation
2) Feedwater (including extraction steam)
3) Main steam

4) Turbine bypass (condenser vacuum)

These systems or their equivalents are the same ones which potentially cause problems in PWR's. Because of their large capacities, these systems, along with the systems which control or support them, are the key non-1E systems which could affect plant safety.

The Chapter 15 analysis evaluated the worst failures of various system components. However, it did not specifically look at combinations of failures resulting from the possible additive effects of these systems. This analysis has determined that due to a control system failure resulting from a harsh environment, the plant can be placed in a condition which has not been previously analyzed.

In the evaluation process, it became clear that the control features of these systems were the key to system interactions. The following control systems are the key contributors and have been evaluated extensively:

1) Recirculation Valve Flow Control System
2) Turbine Driven Feedwater Pump Control System
3) Extraction Steam Valve Control System
4) Steam Bypass and Pressure Regulator System

The effect of these systems on the ability of the safety-related systems to maintain control of reactivity, reactor vessel water level, reactor system pressure, and decay heat removal was evaluated.

The following combinations of adverse system states were analyzed to have the potential to result in reactivity insertions greater than those analyzed in Chapter 15.

Adverse Condition #1

Pipe break at Environmental Zone CT2 could affect the following:

Recirculation Valves	Feedwater Pumps
1B33N011A	1C34N003B
1B33N011B	1C34N003D

Feedwater Pumps - Failure mode unknown.
Recirculation Valves - Failure mode unknown

EVALUATION: Instruments 1B33N011B and 1C34N003D are located in Panel 1H22P041 between the drywell and containment wall on level 620'. The remaining two instruments are in Panel 1H22-P025 on the same level. The only high energy piping in the vicinity of these panels are control rod drive hydraulics.

Perry design ensures that all high-energy lines between the drywell and reactor building wall are restrained from whipping by pipe restraints. Therefore, we may eliminate the effects of whipping pipe (see FSAR 3.6). The control rod drive piping does not carry high temperature water so we can neglect increases in temperature in the area of these transmitters. Therefore, a postulated CRD line break would not affect the operability of the instruments in this area.

Adverse Condition #2

Pipe break in the Heater Bay could affect the following:

Feedwater Pumps	Extraction Steam Valves
1N27N156A	1N25N263A
1N27N156B	1N25N263B
1N27N087A	1N25N303A
1N27N087B	1N25N303B
	1N36N030A,B,C

Extraction Steam Valves - Fail closed
Feedwater Pumps - Fail as is

EVALUATION: It was determined that the only high-energy lines located in the vicinity of these instruments were feedwater lines. Instruments 1N25N263A(B) and 1N25N303A(B) are isolated from the other instruments by a concrete wall. Instrument 1N27N156A is also isolated from the other instruments by walls. Therefore, that leaves only the remaining instruments that could cause multiple failures. Instruments 1N27N156B and 1N27N087A(B), which are located in Panel H51-P098, are located approximately forty feet from Panel H51-P1330 which contains 1N36N030A, B, and C. Failure of 1N36N030A, B, C will cause a main turbine trip at the same time the extraction steam valves go closed. Feedwater pumps would fail as is, this would not increase flow to the reactor. This event is bounded by Chapter 15 analysis.

Adverse Condition #3

Pipe break at Environmental Zone TB1 could affect the following:

Recirculation Valves	Extraction Steam
1R23S004	1R23S004
1R22S003	1R23S006
	1R22S003

Feedwater Pumps	Main Steam & Turbine Bypass Valves
1R23S004	1R42S021
1R23S003	1C85N001A,B
1R24S034	1C85N002A,B,C
1R42S017	1R14S008
1R42S022	
1R22S002	
1R22S003	
1R42S023	

EVALUATION: CEI performed a walkdown of these control systems and instruments. All of the R-system components are located in the Turbine-Power

Complex, which does not contain any high-energy lines. Therefore, the possibility of multiple failures due to high-energy line break does not exist.

The following combinations of adverse system states were further analyzed to identify any conditions which could result in level perturbations outside the bounds of Chapter 15:

Adverse Condition #4

Pipe break at Environmental Zone CT2 could affect the following:

Recirculation Valves	Feedwater
1B33N011A,B	1C34N003B,D

EVALUATION: See Adverse Condition #1.

Adverse Condition #5

Pipe break at Environmental Zone TB1 could affect the following:

Recirculation Valves	Turbine Bypass
1R23S004	1R42S021
1R22S003	1C85N001A,B
	1C85N002A,B,C
Recirculation Pumps	1R14S008
1R22S002	
1R22S003	

Feedwater Pumps	Feedwater Booster Pumps
1R23S004	1R11S004
1R23S003	1R11S005
1R24S034	1R22S002
1R42S017	1R22S003
1R42S022	1R22S004
1R42S002	1R22S005
1R42S023	1R22S006
1R42S021	1R22S016
1R14S008	1R22S017
1R22S003	1R42S022
	1R42S023

EVALUATION: Refer to Adverse Condition #3.

CONCLUSION: The results of our analysis shows that multiple control system malfunctions, due to harsh environment caused by a high-energy line break, does not result in consequences more severe than analyzed in Chapter 15 or beyond the capability of operators or safety systems.

QUESTION

420.06

The analyses reported in Chapter 15 of the FSAR are intended to demonstrate the adequacy of safety systems in mitigating anticipated operational occurrences and accidents.

Based on the conservative assumptions made in defining these design-basis events and the detailed review of the analyses by the staff, it is likely that they adequately bound the consequences of single control system failures.

To provide assurance that the design basis event analyses adequately bound other more fundamental credible failures you are requested to provide the following information:

1) Identify those control systems whose failure or malfunction could seriously impact plant safety.

2) Indicate which, if any, of the control systems identified in (1) receive power from common power sources. The power sources considered should include all power sources whose failure or malfunction could lead to failure or malfunction of more than one control system and should extend to the effects of cascading power losses due to the failure of higher level distribution panels and load centers.

3) Indicate which, if any, of the control systems identified in (1) receive input signals from common sensors. The sensors considered should include, but should not necessarily be limited to, common hydraulic headers or impulse lines feeding pressure, temperature, level or other signals to two or more control systems.

4) Provide justification that any simultaneous malfunctions of the control systems identified in (2) and (3) resulting from failures or malfunctions of the applicable common power source or sensor are bounded by the analyses in Chapter 15 and would not require action or response beyond the capability of operators or safety systems.

APPROACH

To evaluate this type of interaction, CEI has 1) identified those functions which are required to maintain the reactor in a safe condition; 2) identified all systems (both safety and nonsafety) which are connected to and could potentially affect the reactor; 3) determined which of those systems identified in Step 2 that could significantly impact the required safety functions. If there appeared to be a system or combination of systems which could significantly impact the required safety functions, an investigation was made to determine whether any control and electrical dependencies existed which could cause interactions among these systems and thereby effect their function.

The functions which require control to maintain the reactor in a safe condition are: reactivity, reactor vessel level, reactor system pressure, and decay heat removal.

In order to identify those systems which impact these functions, CEI developed a control volume of the reactor vessel, showing all the systems which are connected to it. Since the PNPP has two safety-related redundant trains powered by 1E qualified systems, only interactions between nonsafety and safety systems resulting from control or electrical failures in the nonsafety systems and propagating through the fluid process were considered.

The Class 1E systems which do not require further evaluation for electrical or instrumentation control failures are:

1) Automatic depressurization
2) Reactor core isolation cooling
3) Residual heat removal
4) High pressure core spray
5) Low pressure core spray
6) Control rod drive (scram portion)
7) Nuclear boiler instrumentation

For nonsafety systems, there is a spectrum of possible flow rates to/from the reactor vessel ranging from a few gpm for the sampling system to tens of thousands of gpm for the feedwater system. The systems with the largest capacity will cause the quickest and greatest potential impact on the required safety functions. Each system was evaluated to determine whether the system capacity warranted further analysis. The results of the evaluation are:

1) The standby liquid control system inserts negative reactivity into the core. The insertion of negative reactivity is always a conservative step in an accident or transient analysis. The SLC system cannot remove water; it is a manually initiated system which is strictly a backup to a redundant safety-related system. Its failure or malfunction will have no adverse impact on plant safety.

2) The control rod drive system is required to assure scram capability. The components required to scram the reactor are redundant and Class 1E. The control rod drive pumps are non-1E and pump a relatively small quantity of fluid into the reactor when compared with the feedwater system. Any failure of the non-1E portions will have no adverse impact on plant safety.

3) The reactor plant sampling system is a low capacity system which is manually controlled. The system is automatically isolated by redundant Class 1E isolation valves. There are no non-1E failures associated with this system which could have an adverse impact on plant safety.

4) The reactor water cleanup system is normally operating system with minimal capacity compared to the feedwater system. The system is automatically isolated by redundant Class 1E isolation valves on low reactor water level. The failure or malfunction of non-1E portions of this system will not seriously impact plant safety.

5) The head vent system valves are manually operated. Valve position is not affected by a bus failure.

The conservatisms in the design of the safety-related systems are adequate to absorb the small system effects from the above systems. The systems remaining to be evaluated are:

1) Recirculation
2) Feedwater (including extraction steam)
3) Main steam
4) Turbine bypass (condenser vacuum)

It is noted that these systems or their equivalents are the same ones which potentially cause problems in PWR's. Because of their large capacities, these systems, along with the subsystems which control them, are the key non-1E systems which could affect plant safety.

The importance of these systems is evident from the accidents which have already been analyzed in Chapter 15. The worst nonbreak accidents involve failures which cause these systems to malfunction.

However, the Chapter 15 analysis did not specifically look at combinations of failures resulting from the possible additive effects of these systems. This analysis has determined if the plant can be placed in a condition which as not been analyzed previously.

In the evaluation process, it became clear that the control features of the systems were the key to system interactions. The following control systems are the key contributors and have been evaluated extensively:

1) Recirculation valve flow control system
2) Steam bypass and pressure regulator system
3) Turbine driven feedwater pump control system
4) Extraction steam valve control system

The result of that analysis are presented in the response to Question 402.06 given below.

The following adverse system alignments were analyzed for conditions which could result in a transient not analyzed in FSAR Chapter 15 analysis.

Adverse Condition #1

Failure or 13.8kV bus L-10, 1R22-S001.

EVALUATION: Loss of this bus has the potential of causing a main turbine trip due to the circulation pump C loss and its subsequent effect on condenser vacuum. This main turbine trip without additional complications is bounded by Chapter 15 load rejection analysis.

Adverse Condition #2

Failure of 13.8kV Bus L11, 1R22S002, affect the following:

Recirculation Pump - Trip
Feedwater Pumps - Trip
Feedwater Booster Pumps - Trip

The postulated scenario, inadvertent recirculation pump trip and concurrent reduced feedwater flow, is bounded by FSAR Chapter 15 analysis. Normal water level will decrease to Level 3 at which time scram will be initiated. After that, RCIC and HPCS will maintain water level.

Loss of this bus causes rundown of recirculation pump A and main turbine trip on low condenser vacuum due to loss of offgas refrigerators and circulation pump A. Since partial loss of recirculation flow would immediately start reducing reactor power, an immediate or delayed turbine trip would produce an equal or less severe transient than the load rejection event of Chapter 15. Therefore, this event is bounded.

Adverse Condition #3

Failure of 13.8kV Bus L12, 1R22S003, affects the following:

Recirculation Pump - Trip

Recirculation Valves - Lock in place

Feedwater Pumps - Trip

Feedwater Booster Pumps - Trip

Extraction Steam - Valves Close

EVALUATION: Loss of this bus or associated lower busses will produce some or all of the following effects: Immediate main turbine trip and reduction in feedwater temperature, reduction in feedwater flow, recirculation flow decrease. Of these, the only effect capable of causing a new positive reactivity insertion is the loss of feedwater heating.

Failure of bus 1R25-S012, ancillary to bus 1R22-S003, results in simultaneous main turbine trip and start of feedwater temperature reduction. This event, immediate main turbine trip with start of feedwater temperature reduction, does not result in consequences more severe than the load rejection transient event of Chapter 15. Therefore, this event is bounded.

Adverse Condition #4

Failure of 480V Bus F-1-D, 1R23S004, affects the following:

Feedwater Pumps - Trip

Recirculation Valves - Lock in place

Extraction Steam Valves - Fail closed

EVALUATION: There is no reactivity addition since there is no feedwater flow and recirculation flow is constant. Transient covered by Chapter 15 analysis.

Adverse Condition #5

Failure of the Unit Auxiliary Transformer, 1S11S003, affects the following:

Feedwater Pumps - Trip
Extraction Steam Valves - Close
Recirculation Valves - Lock in place
Recirculation Pumps - Trip

EVALUATION: In this state, the known failure mode of the extraction steam valves causes an increase in reactivity as a result of colder feedwater. If the feedwater pump flow increases and/or recirculation flow increases, we have a transient not analyzed in Chapter 15.

However, the failure of thi bus also trips the steam supply valves to one of the feedwater pumps and trips the feedwater booster pumps. The net result is a loss of feedwater which means that the net adverse alignment, assuming the least favorable results, is an increase in recirculation flow. A recirculation pump also trips. The postulated scenario, inadvertent recirculation pump trip and concurrent reduced feedwater flow, is bounded by FSAR Chapter 15 analysis. Normal water level will decrease to Level 3 at which time scram will be initiated. After that, RCIC and HPCS will maintain water level.

Adverse Condition #6

Failure of 480V Distribution Panel F1D12, 1R25S004, affects the following:

Recirculation Valves - Lock in place
Extraction Steam Valves - Closed

EVALUATION: There is some reactivity addition from colder feedwater, but this addition is less severe than the reactivity addition analyzed in Chapter 15.

Adverse Condition #7

Failure of Control Complex 120 VAC Miscellaneous Distribution Panel K-1-R, 1R25S122, affects the following:

Recirculation Valves - Lock in place
Extraction Steam Valves - Closed

EVALUATION: There is some reactivity addition from colder feedwater, but this addition is less severe than the reactivity addition analyzed in Chapter 15.

Adverse Condition #8

Failure of Static Transfer Switch, 1R14S008, or station normal 125V, Battery 1A, 1R42S001, or 125V DC Bus D-1-A, 1R42S021, affects the following:

Turbine Bypass Valves - No effect
Feedwater Pumps - Increase flow

EVALUATION: Loss of this bus will result in an increase in feedwater flow and subsequent water level 8 trip causing main turbine trip. This event is similar and bounded by the feedwater runout event analyzed in Chapter 15.

CONCLUSION: The results of our analysis shows that the failures of power sources, sensors, or sensor impulse lines, which provide power or signals to two or more control systems, will not result in consequences outside the bounds of Chapter 15 analysis or beyond the capability of operators or safety systems.

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