Mrs. C. Tilton
2443 Warrenville Road
Suite 210
Lisle, Illinois 60532-4352

W. Hodge
500 Circle Dr.
Buchanan, MI 49107

Re: DC Cook CDBI response to question 2012-CDBI-298

Dear Mrs. Tilton,

Attached is the white paper supporting the DC Cook licensing basis position on Steam Generator Tube Rupture. This white paper is intended to address the following potential violations discussed in the NRC technical exit on August 23, 2012.

- Potential violation of T.S. 5.4.1 for emergency response with loss of control air during SGTR. The NRC contends that steps in procedures should have accounted for the time required to put nitrogen in service to provide motive force to the SG PORVs, as well as contending we cannot credit the CAC. If we truly needed Nitrogen, via the procedure it would not have been lined up in time to support the safety analysis.

- Potential violation for non-conservative T.S 3.7.4 for Control Air Compressor out of service. The NRC has a concern that we may have potentially exceeded our allowed outage time on SG PORVs (assuming CAC out of service would make the SG PORVs inoperable) or not having adequate compensatory actions for equipment inoperable.

Sincerely,

W. Hodge

DC Cook Steam Generator Tube Rupture Margin to Overfill Licensing Basis Position

1. Background

During the DC Cook 2012 CDBI inspection the NRC inspectors questioned the design and licensing basis for Steam Generator Tube Rupture (SGTR) Margin to Overfill (MTO). Specifically the design of the compressed air system as a motive force for the Steam Generator Power Operated Relief Valves, with regard to reliance on non-safety grade equipment to mitigate an accident, and the ability to supply compressed air with equipment out of service, and a concurrent loss of offsite power in the affected unit.

2. DC Cook Licensing History Discussion

DC Cook's original FSAR made the assumption that S/G overfill would not occur because of the expectation that the primary-secondary pressure differential could be eliminated within 30 minutes. As a result, the original SGTR only examined the dose consequences from the accident, assuming a loss of condenser due to a loss of offsite power or loss of condensate system. The loss of condenser forced use of the S/G PORVs to cool down the plant. No further examination of the LOOP or its consequences to other equipment, other than the reliance on the SG PORVs, was considered. The assumption stated in the early FSAR, e.g., in Amendment 75 of the Unit 2 section of 14.2.4 of April 1977, is that "Assuming normal operation of the various plant control systems, the following sequence of events is initiated by a tube rupture…" This supports the position that the LOOP was not used in the original accident analysis, other than to force discharge of the steam from the affected S/G to the environment instead of the condenser.

This is further supported by a note in Section 15.1 of the original DC Cook SER that indicates SGTR was only viewed from the prospect of radiological releases to the environment.

In 2000, Westinghouse completed a detailed SGTR analysis for each unit to determine if the assumption that the steam generator did not overfill remained valid and to verify that the offsite release mass remained bounding. Other than the more sophisticated analysis tool (the LOFTTR2 computer code) that was employed, after receiving NRC approval for its use, no changes were made to the Cook Plant SGTR licensing basis.

Following the approval of the SER for the supplemental SGTR MTO methodology in October of 2001 (Attachment 1), the methodology detailed in the License Amendment Request documented in C1000-11 (Attachment 2) and RAI responses documented in C0601-21 (Attachment 3), was added to the DC Cook UFSAR in November of 2001. The SER approving SGTR MTO for DC Cook, explicitly states that no single failure is considered.

The Unit 1 and Unit 2 MUR license amendments further support the acceptance by the Staff on the positions stated above. Cook Plant's response to NRC Question No. 4 in AEP:NRC:2900-01 and the NRC's SER for Unit 1 MUR on December 20, 2002 (see Section 3.2.2.3.3 of the SER) demonstrate that NRC accepts our use of nominal conditions (i.e. the unit is operating with equipment in normal alignment, and all equipment is available) in the analyses.

This position conforms with the majority of nuclear stations licensed in the 1960's and 1970's. The two DC Cook units received their operating licenses in 1974 and 1977. Our research undertaken to understand the requirement for a LOOP/SGTR and single failure showed that NUREG-0800 was updated in the 1996 timeframe and added an additional design basis feature to include the most limiting single failure to the LOOP/SGTR analysis. D C Cook did not receive a backfit order to adopt the new position contained in the 1996 SRP NUREG-0800.

3. Licensing Basis Discussion

The current DC Cook licensing basis SGTR MTO calculation is a best-estimate analysis performed using the LOFTTR2 code, and assumes a loss of offsite power, resulting in the loss of the condenser, but does not assume a single failure.

As called out in Letter C0601-21, "control air appurtenances" along with the SG PORVs are non-safety related and relied upon to mitigate a SGTR. Section 4 of this paper describes in further detail from the UFSAR all the components which make up the control air system, which includes the normal source of compressed air (the two Plant Air Compressors), the backup source of compressed air (the Control Air Compressor), as well as the associated piping, valves, etc. Note that the Nitrogen system is not included as part of the described control air system, it can be used to provide motive force to the SG PORVs but is only included as part of a defense in depth design. Since the normal and backup sources of compressed air are included as part of control air appurtenances these components are assumed to be available to mitigate a SGTR.

The SGTR SER approves mitigation of the SGTR event without the consideration of a single failure. Since the Plant Air Compressors and Control Air Compressor are assumed to be available, and no single failure is required to be considered; these components are allowed to be used in the mitigation of an SGTR. The LOFTTR2 code which was used to analyze the SGTR transient does not model the control air system explicitly, the control air system is assumed to be available in the model to provide motive force to the Steam Generator PORVs, and as documented in the SGTR SER:

> *"NRC staff concludes that the licensee can incorporate the LOFTTR2 code into its licensing bases for CNP and can use the LOFTTR2 code, with the current licensing basis assumptions as inputs for the overfill analysis of steam generator tube rupture accidents."*

AEP letter C0601-21 clearly states that the control air to the SG PORVs is non-safety grade. The SGTR SER states that the NRC reviewed the UFSAR and concluded that the licensing basis SGTR does credit limited use of non-safety grade equipment for mitigating the SGTR. The NRC acceptance of the use of non-safety grade control air to mitigate an SGTR would seem to imply a lack of expectation that the control air system be provided with emergency onsite power in the event of a loss of offsite power. Therefore the assumption that compressed air would be available to mitigate the SGTR event, and compressed air can be provided from either the normal source of compressed air (Plant Air Compressors) or the backup source of compressed air (Control Air Compressor) is in compliance with the DC Cook licensing basis.

4. Compressed Air System Description

Unit 1 is licensed independently of Unit 2, and is considered to be separate with the exception of explicitly identified shared systems defined in the UFSAR. D.C. Cooks UFSAR section 1.3.9.h (Attachment 6) identifies that the compressed air system is a shared system. As described in UFSAR chapter 9.8.2 (Attachment 7) the compressed air system is required to provide a continuous supply of compressed air to vital loads. Attachment 4 of this letter contains system overview drawings, which supplement this discussion, showing the arrangement and interconnection of the compressed air system. As described in UFSAR 9.8.2.3 during normal operation a single Plant Air Compressor (PAC) supplies both units' plant and control air headers. In the event of low plant air header pressure the second PAC will automatically start and load. A low control air header pressure in either unit will automatically start the Control Air Compressor (CAC) in that respective unit. The compressed air system is designed such that the CAC for each unit is supplied by an emergency power supply. If the CAC is out of service for any reason during a SGTR event, compressed air would be available from the Plant Air Compressor powered by the unaffected unit.

While not controlled by Technical Specifications DC Cook is sensitive to the importance of the compressed air system. The unavailability of control air system is controlled by the maintenance rule. Specifically, functions CA-02 and MS-05 in Maintenance Rule (10 CFR 50.65) Scoping Documents for the Compressed Air and Main Steam Systems, respectively. Additionally work control procedure PMP-2291-WMP-001 (Attachment 8) requires both PACs to be guarded in the event a CAC is out of service; or to guard the opposite unit PAC, both CACs, and the Backup Air Compressor if a PAC is taken out of service.

5. Electrical Power Description

The scenario postulated by the inspectors involves the loss of offsite power to the affected unit. Since the compressed air system is shared between units and the normal supply of control air may be supplied from either unit, it is important to consider the sources of power to the air compressors.

The following is a detailed discussion addressing specifically how AC power is available to the maintained plant air system. The information below is supplemented by system overview drawings contained in Attachment 5.

Normally, auxiliary power is supplied via the Unit Aux Transformers 1AB(2AB) and 1CD(2CD) powered via the associated unit's Main Generator. A loss of offsite power to the affected unit's Plant Air Compressor is likely to result in a loss of the Technical Specification related source of offsite power to the unaffected unit's Plant Air Compressor. The loss of Technical Specification source of offsite power to the unaffected unit does not result in a loss of power to the unaffected unit's Plant Air Compressor. The unaffected unit will remain in operation with power supplied to the Plant Air Compressor and its Aux Oil Pump via the Unit Aux Transformers.

6. Licensing Implications of the Postulated Scenario

DC Cook provided the list of equipment required to mitigate a SGTR to the NRC in the RAI response documented in C0601-21. Included in this list is non-safety-related control air appurtenances, which includes the normal and backup supplies of control air. Since the normal and backup sources of control air are the Plant Air Compressors and the Control Air Compressor respectively, and the NRC implicitly approved the use of this equipment to mitigate a SGTR without the consideration of a single failure. Both the normal and backup supplies of air are considered valid sources of compressed air to mitigate a SGTR and coincident loss of offsite power. The normal supply of control air is shared between units, and based on the discussion in Section 5 a loss of offsite power in Unit 1 would not result in a loss of the Plant Air Compressor in Unit 2, and vice versa.

In the postulated scenario where a Control Air Compressor is out of service, and that unit experiences a SGTR with a concurrent loss of offsite power; per the DC Cook design and licensing basis the unaffected unit Plant Air Compressor is a valid supply of compressed air for the mitigation of the SGTR event. Because no single failure is considered in the DC Cook licensing basis, the unaffected unit Plant Air Compressor is assumed to be available and functional. If the postulated scenario precludes the use of compressed air supplied from the unaffected unit Plant Air Compressor, relative to the DC Cook licensing basis that would be considered a single failure, and considered beyond the DC Cook design basis.

7. Conclusion

DC Cook Units 1 and 2 were licensed and have as a design basis the ability to address the SGTR coincident with a Loss of Offsite Power. Inclusion of another single failure is a beyond design basis event for D C Cook. As the plant air system is described in the UFSAR as a shared system and assumed to be available in the SGTR with LOOP analysis, failure to credit the air system would constitute another single failure. The postulated scenario of a SGTR with dual unit LOOP and the Control Air Compressor out of service, posed by the NRC inspectors is believed to be beyond design and licensing basis, and therefore not subject to evaluation via the inspection or licensing process at this time.

The DC Cook design and licensing basis allows the use of non-safety related control air appurtenances, which include the normal and backup supplies of compressed air. As described in the UFSAR, DC Cook is designed with compressed air as a shared system. In the event of a SGTR with concurrent loss of offsite power in one unit, the unaffected unit can provide power to its Plant Air Compressor. In the event that a Control Air Compressor is out of service and that unit experiences a SGTR with a loss of offsite power, the source of control air would be the unaffected unit Plant Air Compressor. Exclusion of the alternate unit Plant Air Compressor as a valid source of compressed air to mitigate the SGTR event would be considered a single failure, and beyond the current DC Cook design and licensing basis.

Attachment 1

SER Approving the SGTR MTO methodology for DC Cook

October 24, 2001

Mr. Robert P. Powers, Senior Vice President
Indiana Michigan Power Company
Nuclear Generation Group
500 Circle Drive
Buchanan, MI 49107

SUBJECT: DONALD C. COOK NUCLEAR PLANT, UNITS 1 AND 2 - ISSUANCE OF AMENDMENTS (TAC NOS. MB0739 AND MB0740)

Dear Mr. Powers:

The U.S. Nuclear Regulatory Commission has issued the enclosed Amendment No. 256 to Facility Operating License No. DPR-58 and Amendment No. 239 to Facility Operating License No. DPR-74 for the Donald C. Cook Nuclear Plant, Units 1 and 2. The amendments are in response to your application dated October 24, 2000, as supplemented June 29, 2001, approve changes to the updated final safety analysis report (UFSAR).

The amendments would approve changes to the UFSAR to incorporate a supplemental methodology for the analysis of steam generator overfill following a steam generator tube rupture.

A copy of our related safety evaluation is also enclosed. A Notice of Issuance will be included in the Commission's next biweekly *Federal Register* notice.

Sincerely,

/RA/

John F. Stang, Senior Project Manager, Section 1
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket Nos. 50-315 and 50-316

Enclosures: 1. Amendment No. 256 to DPR-58
2. Amendment No. 239 to DPR-74
3. Safety Evaluation

cc w/encls: See next page

October 24, 2001

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Indiana Michigan Power Company
Nuclear Generation Group
500 Circle Drive
Buchanan, MI 49107

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Docket Nos. 50-315 and 50-316

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cc w/encls: See next page

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Donald C. Cook Nuclear Plant, Units 1 and 2

cc:

Regional Administrator, Region III
U.S. Nuclear Regulatory Commission
801 Warrenville Road
Lisle, IL 60532-4351

Attorney General
Department of Attorney General
525 West Ottawa Street
Lansing, MI 48913

Township Supervisor
Lake Township Hall
P.O. Box 818
Bridgman, MI 49106

U.S. Nuclear Regulatory Commission
Resident Inspector's Office
7700 Red Arrow Highway
Stevensville, MI 49127

David W. Jenkins, Esquire
Indiana Michigan Power Company
Nuclear Generation Group
One Cook Place
Bridgman, MI 49106

Mayor, City of Bridgman
P.O. Box 366
Bridgman, MI 49106

Special Assistant to the Governor
Room 1 - State Capitol
Lansing, MI 48909

Drinking Water and Radiological
  Protection Division
Michigan Department of
  Environmental Quality
3423 N. Martin Luther King Jr Blvd
P.O. Box 30630, CPH Mailroom
Lansing, MI 48909-8130

Ronald Gaston
Director, Regulatory Affairs
Indiana Michigan Power Company
Nuclear Generation Group
One Cook Place
Bridgman, MI 49106

David A. Lochbaum
Union of Concerned Scientists
1616 P Street NW, Suite 310
Washington, DC 20036-1495

A. Christopher Bakken, Site Vice President
Indiana Michigan Power Company
Nuclear Generation Group
One Cook Place
Bridgman, MI 49106

Michael W. Rencheck
Vice President, Nuclear Engineering
Indiana Michigan Power Company
Nuclear Generation Group
500 Circle Drive
Buchanan, MI 49107

INDIANA MICHIGAN POWER COMPANY

DOCKET NO. 50-315

DONALD C. COOK NUCLEAR PLANT, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 256
License No. DPR-58

1. The U.S. Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Indiana Michigan Power Company (the licensee) dated October 24, 2000, as supplemented June 29, 2001, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, by Amendment No. 256, Facility Operating License No. DPR-58 is hereby amended to authorize a change to the methodology used in the evaluation of a steam generator (SG) overfill following a SG tube rupture referenced in the updated final safety analysis report (UFSAR), as set forth in the license amendment application dated October 24, 2000, as supplemented June 29, 2001, and evaluated in the associated safety evaluation by the Commission's Office of Nuclear Reactor Regulation. The licensee shall update the UFSAR by adding a description of this change, as authorized by this amendment, and in accordance with 10 CFR 50.71(e).

3. This license amendment is effective as of its date of issuance and shall be implemented within 30 days.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

William D. Reckley, Acting Chief, Section 1
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Date of Issuance: October 24, 2001

INDIANA MICHIGAN POWER COMPANY

DOCKET NO. 50-316

DONALD C. COOK NUCLEAR PLANT, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 239
License No. DPR-74

1. The U.S. Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Indiana Michigan Power Company (the licensee) dated October 24, 2000, as supplemented June 29, 2001, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, by Amendment No. 239, Facility Operating License No. DPR-74 is hereby amended to authorize a change to the methodology used in the evaluation of a steam generator (SG) overfill following a SG tube rupture referenced in the updated final safety analysis report (UFSAR), as set forth in the license amendment application dated October 24, 2000, as supplemented June 29, 2001, and evaluated in the associated safety evaluation by the Commission's Office of Nuclear Reactor Regulation. The licensee shall update the UFSAR by adding a description of this change, as authorized by this amendment, and in accordance with 10 CFR 50.71(e).

3. This license amendment is effective as of its date of issuance and shall be implemented within 30 days.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

William D. Reckley, Acting Chief, Section 1
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Date of Issuance: October 24, 2001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 256 TO FACILITY OPERATING LICENSE NO. DPR-58

AND AMENDMENT NO. 239 TO FACILITY OPERATING LICENSE NO. DPR-74

INDIANA MICHIGAN POWER COMPANY

DONALD C. COOK NUCLEAR PLANT, UNITS 1 AND 2

DOCKET NOS. 50-315 AND 50-316

1.0 INTRODUCTION

By application dated October 24, 2000, as supplemented June 29, 2001, the Indiana Michigan Power Company (the licensee) requested approval of changes to the updated final safety analysis report (UFSAR). The proposed amendments would approve incorporating a supplemental methodology into the analysis of steam generator (SG) overfill following a steam generator tube rupture (SGTR). The new analysis more realistically models the operator actions and plant response. The results from the calculations would then be used to determine the time available for operator actions to prevent overfill and to revise the plant Emergency Operating Procedures.

The current SGTR analysis for D. C. Cook Nuclear Plant (CNP) Units 1 and 2, calculates an average break flow from the time of the accident until the reactor trips and safety injection (SI) initiates. Following SI initiation, the UFSAR analysis uses an equilibrium break flow that continues at a constant rate for 30 minutes. The resulting break flow mass transfer is then used to calculate the radiological consequences of the SGTR. The UFSAR calculations are considered conservative since the reduction in the break flow rate over the thirty minute period is ignored. Inherent in this evaluation is the assumption that the operator can terminate the break flow in 30 minutes, and that the termination will prevent the SG from overfilling. The 30 minute time to termination is the current licensing basis referenced in the UFSAR.

During an expanded system readiness review, the licensee conducted SGTR simulator exercises, and the operators demonstrated that the time to terminate the tube rupture break flow exceeded the 30 minute assumption. Should operators significantly exceed the 30 minute termination criteria, the SG could overfill, assuming the break flow documented in the UFSAR occurred. Upon SG overfill, water enters the steam lines. The steam lines at D. C. Cook are not qualified to contain water, and the pressure operated relief valves (PORVS) are not qualified to relieve water. This event could have radiological consequences beyond the design basis of the D. C. Cook Nuclear Plants.

The licensee proposes to incorporate a supplemental methodology into the analysis of steam generator overfill following a SGTR. The licensee plans to use the results from the analysis to determine the time available for operator actions to prevent overfill and to revise the plant Emergency Operating Procedures.

The licensee’ s June 29, 2001, letter, provided additional information only in response to the Nuclear Regulatory Commission (NRC) staff's request for additional information dated May 7, 2001. The letter provided clarifying information within the scope of the original application and did not change the initial proposed no significant hazards consideration determination.

2.0 EVALUATION

The discrepancy between the simulator exercises and the assumptions in the SGTR analysis, required the licensee to reexamine the SGTR accidents. The licensee proposes implementing WCAP-10698-P-A “SGTR Analysis Methodology to Determine the Margin to Steam Generator Overfill,” to update the analysis. To implement the WCAP, the licensee used the LOFTTR2 computer code and the plant-specific current licensing basis assumptions. The improvements in the analysis included modeling operator actions, enhancing the SG secondary side model, and improving the tube rupture break flow model.

From the new analysis, the licensee plans to determine the time available for operator actions to prevent steam generator overfill. They will then correct the discrepancy between the UFSAR analysis and the demonstrated plant response with regard to SGTR overfill. Based upon the results of the calculations, the licensee also plans to update the emergency operating procedures to incorporate the enhancements identified by the new analysis.

The licensee's use of WCAP-10698-P-A presented several exceptions from the requirements of the NRC staff's safety evaluation that approved it. These exceptions are presented in more detail below as Items 1 through 5, and are summarized in Table 1. Overall, WCAP-10698-P-A describes a conservative methodology for SGTR overfill calculations. The WCAP applies the LOFTTR1 computer code for modeling the SGTR transient. Input assumptions into the accident analysis model include loss of offsite power, most reactive rod stuck, conservative initial conditions, turbine runback, 120 percent of 1971 ANS (American Nuclear Society) decay heat standard, and the worst single failure, among other assumptions. However, variations in plant designs prevent a single model from adequately representing all Westinghouse plants. Because of these variations, the NRC required that plant-specific input be provided when utilities reference the WCAP methodology. The required plant-specific input is as follows:

1. Each utility in the SGTR subgroup must confirm that they have in place simulators and training programs which provide the required assurance that the necessary actions and times can be taken consistent with those assumed for the WCAP-10698 design-basis analysis. Demonstration runs should be performed to show that the accident can be mitigated within a period of time compatible with overfill prevention, using design-basis assumptions regarding available equipment, and to demonstrate that the operator action times assumed in the analysis are realistic.

The licensee confirmed that they have simulator and training programs in place that assure the necessary actions and times can be taken to prevent steam generator overfill. The licensee also stated that the operator action times assumed in the analysis are realistic because they were confirmed by simulator demonstrations. They incorporated these action times into their training programs. However, when establishing the bases for the SGTR transient, the licensee did not assume the same design-basis assumptions as the WCAP-10698 analysis.

2. A site specific SGTR radiation offsite consequence analysis which assumes the most severe failure identified in WCAP-10698, Supplement 1. The analysis should be performed using the methodology in SRP Section 15.6.3, as supplemented by the guidance in Reference (1) - (Note: Reference (1) of the WCAP-10698 Safety Evaluation).

In response to this requirement, the licensee did not submit the offsite consequence analysis. They, instead, proposed keeping their current licensing basis SGTR calculation for the offsite radiological consequences. The licensee states that this proposal is adequate, since the original licensing basis analysis has a greater primary to secondary side mass transfer than the analysis performed using the WCAP methodology.

3. An evaluation of the structural adequacy of the main steam lines and associated supports under water-filled conditions as a result of SGTR overfill.

Because licensee's evaluation determined that the steam generators do not overfill, the licensee did not submit a structural evaluation of the main steam lines and associated supports for the water-filled conditions of a SGTR overfill incident.

4. A list of systems, components and instrumentation which are credited for accident mitigation in the plant-specific SGTR EOP(s). Specify whether each system and component specified is safety grade. For primary and secondary PORVs and control valves, specify the valve motive power and state whether the motive power and valve controls are safety grade. For non-safety grade systems and components, state whether safety grade backups are available which can be expected to function or provide the desired information within a time period compatible with prevention of SGTR overfill or justify that non-safety grade components can be utilized for the design-basis event. Provide a list of all radiation monitors that could be utilized for identification of the accident and the ruptured steam generator and specify the quality and reliability of this instrumentation if possible. If the EOPs specify SG sampling as a means of ruptured SG identification, provide the expected time period for obtaining the sample results and discuss the effects on the duration of the accident.

The licensee provided the required lists of systems, components, and instrumentation that are used for SGTR accident mitigation. They also specified the safety classification of the systems and power sources. However, the licensee listed several systems used for SGTR mitigation that are not safety related and do not have safety related backups. The licensee justified the use of the non-safety related equipment by stating that these systems are credited in the current UFSAR Section 14.2.4 accident analysis. Upon review of UFSAR Section 14.2.4, the staff concludes that the licensing basis SGTR analysis does credit limited use of non-safety grade equipment for mitigating the SGTR.

5. A survey of plant primary and "balance-of-plant" systems design to determine the compatibility with the bounding plant analysis in WCAP-10698. Major design differences should be noted. The worst single failure should be identified if different from the WCAP-10698 analysis and the effect of the difference on the margin of overfill should be provided.

The licensee did not use the bounding plant analysis of WCAP-10698-P-A, but performed a D. C. Cook plant-specific analysis. Therefore, they did not submit a survey of the plant systems that determined the variance with the WCAP analysis. Also, the licensee did not assume the worst single failure as prescribed by the WCAP-10698-P-A safety analysis, and did not provide it's effect on the margin to overfill. The licensee based their decision not to assume the worst single failure on the fact that their current licensing basis does not include a single failure.

Overall, the proposed D. C. Cook plant methodology and the approved WCAP-10698 methodology vary significantly. The differences are listed in Table 1.

Table 1

WCAP-10698-P-A	**D. C. Cook**
Uses LOFTTR1 computer model	Uses LOFTTR2 computer model
Assumes a worst single failure for design basis	Does not assume worst single failure
Requires offsite radiological consequence analysis, assuming worst failure	Does not provide analysis based on WCAP methodology or worst failure
Requires structural adequacy analysis of SGTR overfill	Does not provide structural analysis
Requires comparison of BOP equipment with WCAP case	Does not provide comparison

Because of the significance of these variations from the WCAP-10698-P-A methodology, the staff cannot endorse the licensee's reference to WCAP-10698-P-A in the licensing basis for a SGTR overfill analysis. However, the LOFTTR2 computer code has been previously accepted for licensing basis SGTR analyses. Therefore, we find it acceptable for the licensee to use LOFTTR2 for their licensing basis SGTR overfill analysis as well as the licensing calculation for the design basis SGTR accident.

Finally, the licensee committed to revise the Unit 1 emergency operating procedures to incorporate enhancements from the new analysis. The licensee has updated the Unit 1and 2 emergency operating procedures. The licensee will revise the UFSAR to incorporate the use of the new methodology in the SGTR analyses. Incorporating the information from the new SGTR model is acceptable. However, as stated earlier, citing WCAP-10698-P-A as the licensing basis for the SGTR overfill analysis is not acceptable.

3.0 SUMMARY

The licensee's method of implementation of WCAP-10698-P-A does not satisfy the WCAP safety evaluation requirements. Therefore, the incorporation of WCAP-10698-P-A into D. C. Cook 's licensing basis is not acceptable as proposed in the October 24, 2000, application. However, the LOFTTR2 computer code has been previously accepted for use in licensing basis analyses. Therefore, the NRC staff concludes that the licensee can incorporate the LOFTTR2 code into its licensing bases for CNP and can use the LOFTTR2 code, with the current licensing basis assumptions as inputs for the overfill analysis of steam generator tube rupture accidents. This change to the licensing basis does not affect accident initiators or precursors. The change also does not increase the probability of an accident or decrease the ability of the operators to mitigate the consequences of an accident. Therefore, the proposed change to the UFSAR is acceptable.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Michigan State official was notified of the proposed issuance of the amendments. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

These amendments change the requirements with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The staff has determined that the amendments involve no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendments involve no significant hazards consideration and there has been no public comment on such finding (66 FR 7682). Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendments.

6.0 CONCLUSION

The staff has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendments will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributor: Sean Peters

Date: October 24, 2001

Attachment 2

DC Cook License Amendment Request for SGTR MTO

Indiana Michigan
Power Company
500 Circle Drive
Buchanan, MI 49107 1373

INDIANA
MICHIGAN
POWER

October 24, 2000

C1000-11
10 CFR 50.90

Docket Nos.: 50-315
50-316

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Mail Stop O-P1-17
Washington, DC 20555-0001

Donald C. Cook Nuclear Plant Units 1 and 2
LICENSE AMENDMENT FOR CHANGES IN STEAM
GENERATOR TUBE RUPTURE ANALYSIS METHODOLOGY

Pursuant to 10 CFR 50.90, Indiana Michigan Power Company, the Licensee for Donald C. Cook Nuclear Plant (CNP) Units 1 and 2, proposes to amend Facility Operating Licenses DPR-58 and DPR-74 to incorporate a supplemental methodology into its analysis of steam generator (SG) overfill following a steam generator tube rupture (SGTR). The proposed change would use the analysis methodology documented in the Westinghouse Electric Company WCAP-10698-P-A, "SGTR Analysis Methodology to Determine Margin to Steam Generator Overfill," to more accurately calculate the transient response of CNP to a postulated SGTR with respect to CNP SG overfill. The break flow model used in WCAP-10698-P-A has been reviewed and accepted by the Nuclear Regulatory Commission (NRC) in safety evaluation report, "Acceptance for Referencing of Licensing Topical Report WCAP-10698, 'SGTR Analysis Methodology to Determine Margin to Steam Generator Overfill,' December 1984," dated March 30, 1987. Since this methodology is not currently part of the CNP licensing basis, use of this methodology for the supplemental SGTR analysis has been determined to involve an unreviewed safety question. Therefore, NRC staff review and approval are required for the addition of this methodology to the CNP licensing basis in accordance with 10 CFR 50.90. The Updated Final Safety Analysis Report will be revised to incorporate the use of the new methodology.

Attachment 1 to this letter provides a detailed description of the proposed changes. Attachment 2 describes the evaluation performed in accordance with 10 CFR 50.92(c), which concludes that no significant hazard is involved.

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Attachment 3 provides the environmental assessment and Attachment 4 provides the new commitments made in this letter.

Copies of this letter and its attachments are being transmitted to the Michigan Public Service Commission and Michigan Department of Environmental Quality, in accordance with the requirements of 10 CFR 50.91.

Should you have any questions, please contact Mr. Wayne J. Kropp, Director of Regulatory Affairs, at (616) 697-5056.

Sincerely,

R. P. Powers
Vice President

\dmb

Attachments

c: J. E. Dyer
MDEQ - DW & RPD
NRC Resident Inspector
R. Whale

AFFIRMATION

I, Robert P. Powers, being duly sworn, state that I am Vice President of Indiana Michigan Power Company (I&M), that I am authorized to sign and file this request with the Nuclear Regulatory Commission on behalf of I&M, and that the statements made and the matters set forth herein pertaining to I&M are true and correct to the best of my knowledge, information, and belief.

Indiana Michigan Power Company

R. P. Powers

R. P. Powers
Vice President

SWORN TO AND SUBSCRIBED BEFORE ME

THIS 24th DAY OF October, 2000

Julie Newmiller
Notary Public

My Commission Expires 8-22-04

JULIE E. NEWMILLER
Notary Public, Berrien County, MI
My Commission Expires Aug 22, 2004

ATTACHMENT 1 TO C1000-11

DESCRIPTION AND SAFETY ANALYSIS FOR THE PROPOSED CHANGES

A. Summary of Proposed Changes

Indiana Michigan Power Company (I&M), the Licensee for Donald C. Cook Nuclear Plant (CNP) Units 1 and 2, proposes to amend Facility Operating Licenses DPR-58 and DPR-74 to incorporate a supplemental methodology into its analysis of steam generator (SG) overfill following a steam generator tube rupture (SGTR). The proposed change would use the analysis methodology documented in the Westinghouse Electric Company (Westinghouse) WCAP-10698-P-A, "SGTR Analysis Methodology to Determine Margin to Steam Generator Overfill," to more accurately calculate the transient response of CNP to a postulated SGTR with respect to CNP SG overfill. The break flow model used in WCAP-10698-P-A has been reviewed and accepted by the Nuclear Regulatory Commission (NRC) in safety evaluation report (SER), "Acceptance for Referencing of Licensing Topical Report WCAP-10698, 'SGTR Analysis Methodology to Determine Margin to Steam Generator Overfill,' December 1984," dated March 30, 1987. The use of this methodology for the supplemental SGTR analysis has been determined to involve an unreviewed safety question (USQ). Therefore, NRC staff review and approval are required for the addition of this methodology to the CNP licensing basis in accordance with 10 CFR 50.90. The Updated Final Safety Analysis Report (UFSAR) will be revised to incorporate the use of the new methodology.

The change in methodology would be applied to supplemental analyses for the determination of the time available for operator actions to prevent overfilling the secondary side of the affected SG in response to an SGTR event. The present methodology would be retained for calculating the radiological consequences of a postulated SGTR since the current CNP licensing basis methodology continues to bound the radiological consequences calculated by the new methodology.

B. Description and Bases for the Current Requirements

The CNP licensing basis analysis for SGTR was included in the December 18, 1967, license application. This analysis was retained as part of the Final Safety Analysis Report, Section 14.2.4, "Steam Generator Tube Rupture," as described in the NRC's SER dated September 10, 1973, and remains in Section 14.2.4 of the UFSAR. This analysis calculates an average SGTR break flow for the time period from the postulated accident initiation until reactor trip and safety injection (SI) initiation, which are assumed to occur simultaneously. This analysis assumes an equilibrium break flow following reactor trip and SI initiation that continues at a constant rate for thirty minutes. The resulting break flow mass transfer is then used to calculate the radiological consequences of the postulated accident.

The calculations performed as part of the SGTR analysis methodology are based on the limited industry experience and calculational capabilities that existed when CNP was licensed. The use of a constant mass flow rate for thirty minutes simplified the analysis calculations and was deemed conservative because the reduction in the break flow rate over the thirty-minute duration is ignored. Inherent in the evaluation is the assumption that an operator can terminate the break flow in thirty minutes, and that a thirty-minute termination time will prevent SG overfill from occurring.

C. Description of the Change and the Need for Revision of the Requirement

I&M proposes to adopt as part of the CNP licensing basis a new, NRC-approved analysis method using a different calculational technique that models operator actions to more accurately predict the time to overfill an SG following an SGTR. The current licensing basis analysis for the postulated CNP SGTR accident assumes that the SG experiencing the tube rupture could be isolated within thirty minutes. That estimate was considered reasonable based on what was known when CNP Units 1 and 2 were licensed. Since that time, additional information has been obtained, procedures have been changed, control room protocols have evolved, and better analysis tools have become available for calculating operational transient responses.

Issues related to the SGTR analysis and the time to SG overfill were identified in March 1999, during the CNP expanded system readiness review. To address these issues, I&M personnel conducted SGTR simulator exercises that adhere to the CNP emergency operating procedures (EOPs). These exercises demonstrated that the time for reactor operators to terminate the tube rupture break flow in the affected SG exceeds thirty minutes. Therefore, the analysis assumption that the affected SG is isolated within thirty minutes is invalid.

The methodology used for the analysis of SGTR events cannot be used to determine if the SG overfills since the simplistic modeling does not consider the secondary side liquid volume transient. The methodology uses a mass and energy balance to calculate the primary-to-secondary leakage through the ruptured tube and steam release to the atmosphere. At thirty minutes, the event is assumed to end as a result of the assumption that primary-to-secondary break flow has been terminated. No operator actions are explicitly modeled in the analysis, although the operators are required to perform the necessary steps to terminate the break flow. Thirty minutes is not assumed a criterion for operator action; it is an analysis assumption used to conservatively calculate the total break flow.

Although the thirty-minute timeframe was selected as an analytical assumption, it could result in water entering the main steam line connected to the affected SG. Overfilling of the affected SG is unacceptable because the steam lines are not qualified for water fill and the SG power-operated relief valves are not qualified to relieve water. Overfill during a SGTR may propagate the event into a main steam line break accident coincident with SGTR. The radiological consequences from such a combination of events have not been analyzed as it is outside of the licensing and design basis for CNP.

I&M determined that a new analysis using a different calculational technique that includes modeling operator actions is needed to more accurately identify the time to overfill. I&M proposes to use the break flow model and associated LOFTRR2 computer code described in WCAP-10698-P-A to quantify SG filling response following a postulated SGTR. The use of this break flow model would supplement the present SGTR analysis that did not specifically address SG overfill.

I&M has performed calculations in accordance with the WCAP-10698-P-A methodology. The new analysis calculation, which accounts for timing of operator actions, indicates that the break flow duration is longer than thirty minutes. The incorporation of a new methodology to more accurately characterize SGTRs and subsequent SG overfill timing into the CNP licensing basis is needed to rectify this non-conformance with the UFSAR description of the CNP response to SGTR events. Incorporation of the new methodology into the CNP licensing basis has been determined to constitute a USQ, requiring review and approval by the NRC.

D. Bases for the Proposed Changes

The current SGTR thermal and hydraulic analyses performed for CNP do not incorporate the advantage of the available computer analysis model to determine the dynamic plant behavior following an SGTR accident. Rather, simplified calculations were performed based on the expected SGTR transient response to determine the primary to secondary break flow and the steam release to the atmosphere for use in calculating the offsite radiation exposure due to the event.

Following the R. E. Ginna Nuclear Power Plant SGTR event in January 1982, a new SGTR analysis methodology was developed. The new methodology is based on modeling operator actions for SGTR recovery derived from plant-specific simulator studies, incorporates an improved SG secondary-side model, and includes a more realistic tube rupture break flow model. The NRC approved the revised SGTR analysis methodology in an SER, "Acceptance for Referencing of Licensing Topical Report WCAP-10698, 'SGTR Analysis Methodology to Determine Margin to Steam Generator Overfill,' December 1984," dated March 30, 1987. As required by the NRC as part of this SER, this methodology has subsequently been adopted by the Westinghouse Owners Group subgroup of plants that received operating licenses in this time frame. I&M has determined that the use of this methodology is applicable to CNP.

The current CNP SGTR analysis does not explicitly address the SG filling response due to an SGTR accident. The new methodology, widely adopted by owners of Westinghouse pressurized water reactor plants, is best suited to analyze the SG secondary-side inventory response following an SGTR as opposed to simplified hand calculations. Thus, approval for use of the methodology is being requested to supplement the CNP SGTR analysis. This will allow the SGTR analysis to reflect more accurately the time to overfill of the affected SG in the CNP licensing basis.

The use of this WCAP methodology and associated computer code for break flow modeling more accurately calculates the plant response to an SGTR event. The improved accuracy of the new methodology provides valuable information related to the analysis of operator actions and the associated timing of these actions. The more accurate calculations related to the SGTR overfill transient provide valuable insights for enhancement of the EOPs. These types of improvements have been incorporated into the Unit 2 EOPs and are in progress for the EOPs for Unit 1. Revisions to the appropriate Unit 1 EOPs to incorporate enhancements identified due to the new SGTR SG overfill methodology will be completed prior to restart from the current outage.

E. Discussion of Risk

The proposed change, to adopt a new analytical method to evaluate the effects of an SGTR, does not affect any accident initiators or precursors. As such, the proposed change does not increase the probability of an accident. The proposed change also does not affect the ability of operators to mitigate the consequences of an accident. The WCAP-10698-P-A methodology calculates more accurately the flow from the reactor coolant system to the SG secondary side following a postulated SGTR. The resulting break flow mass transfer is then used to calculate the radiological consequences of the postulated accident. The current licensing basis methodology for calculating the radiological consequences of a postulated SGTR results in mass transfer values that bound those calculated using the new methodology. As such, the existing licensing basis radiological consequence calculations will be retained. Thus, no additional radiological source terms are generated, and the consequences of an accident previously evaluated in the UFSAR will not be increased. Therefore, the incorporation of the new methodology would not result in an increase in risk for CNP.

ATTACHMENT 2 TO C1000-11

NO SIGNIFICANT HAZARDS CONSIDERATION EVALUATION

Indiana Michigan Power Company (I&M) has evaluated this proposed amendment and determined that it does not involve a significant hazard. According to 10 CFR 50.92(c), a proposed amendment to an operating license does not involve a significant hazard if operation of the facility in accordance with the proposed amendment would not:

1. involve a significant increase in the probability of occurrence or consequences of an accident previously evaluated;

2. create the possibility of a new or different kind of accident from any previously evaluated; or

3. involve a significant reduction in a margin of safety.

I&M proposes to incorporate a supplemental methodology into its analysis of steam generator (SG) overfill following a steam generator tube rupture (SGTR). The proposed change would use the analysis methodology documented in the Westinghouse Electric Company WCAP-10698-P-A, "SGTR Analysis Methodology to Determine Margin to Steam Generator Overfill," to more accurately calculate the transient response of Donald C. Cook Nuclear Plant (CNP) to a postulated SGTR with respect to CNP SG overfill. The break flow model used in WCAP-10698-P-A has been reviewed and accepted by the Nuclear Regulatory Commission (NRC) in safety evaluation report, "Acceptance for Referencing of Licensing Topical Report WCAP-10698, 'SGTR Analysis Methodology to Determine Margin to Steam Generator Overfill,' December 1984," dated March 30, 1987. The use of this methodology for the supplemental SGTR analysis has been determined to involve an unreviewed safety question. Therefore, NRC staff review and approval are required for the addition of this methodology to the CNP licensing basis in accordance with 10 CFR 50.90. The Updated Final Safety Analysis Report (UFSAR) will be revised to incorporate the use of the new methodology.

The change in methodology would be applied to supplemental analyses for the determination of the time available for operator actions to prevent overfilling the secondary side of the affected SG in response to an SGTR event. The present methodology would be retained for calculating the radiological consequences of a postulated SGTR since the current CNP licensing basis methodology continues to bound the radiological consequences calculated by the new methodology.

The determination that the criteria set forth in 10 CFR 50.92 are met for this amendment request is indicated below.

1. Does the change involve a significant increase in the probability of occurrence or consequences of an accident previously evaluated?

The proposed change, to adopt a new analytical method to evaluate the effects of an SGTR, does not affect any accident initiators or precursors. As such, the proposed change does not increase the probability of an accident. The proposed change also does not affect the ability of operators to mitigate the consequences of an accident. The proposed change does not impact the design of the affected plant systems such that previously analyzed systems, structures, and components (SSCs) would now be more likely to fail. The changes will not modify plant systems to reduce their design capability during normal operating and accident conditions. The use of the WCAP-10698-P-A methodology to more accurately calculate the flow from the reactor coolant system (RCS) to the SG secondary side following a postulated SGTR does not affect the probability of any analyzed events. The use of the WCAP-10698-P-A methodology does not affect SGTR initiators or precursors. Therefore, incorporating the new methodology does not affect equipment malfunction probability, nor does it affect or create new accident initiators or precursors. Thus, there will be no reduction in the capability of those SSCs in limiting the consequences of previously evaluated accidents.

Additionally, the present methodology for calculating the radiological consequences of a postulated SGTR is conservative when compared with results from the new methodology. As such, the existing licensing basis radiological consequence calculations will be retained. Thus, no additional radiological source terms are generated, and the consequences of an accident previously evaluated in the UFSAR will not be increased. The use of this WCAP methodology and associated computer code for break flow modeling more accurately calculates the plant response to an SGTR event. The improved accuracy of the new methodology provides valuable information related to the analysis of operator actions and the associated timing. Such accurate transient response information enables enhancements to be made to the emergency operating procedures (EOPs).

Therefore, the proposed changes cannot increase the consequences or probability of occurrence of an accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The proposed change does not impact the design of affected plant systems, involve a physical alteration to the systems, or change to the way in which systems are currently operated, such that previously unanalyzed SGTRs would now occur. The change to incorporate the WCAP-10698-P-A methodology does not introduce any new malfunctions; it calculates more accurately the flow from the RCS to the SG secondary side following a postulated SGTR to determine the time available for operator actions to prevent overfilling the affected SG.

Thus, use of the WCAP-10698-P-A methodology does not affect or create new accident initiators or precursors or create the possibility of a new or different kind of accident.

Therefore, the proposed changes do not increase the possibility of a new or different kind of accident than previously evaluated.

3. Does the change involve a significant reduction in a margin of safety?

The approval of the license amendment will not result in any modifications to affected plant systems that would reduce their design capabilities during normal operating and accident conditions. By using the WCAP-10698-P-A methodology, a more accurate SGTR response is calculated. The improved understanding of the transient response enables enhancements to the EOPs, which provide further assurance that SSCs required for accident mitigation are protected.

Therefore, the proposed changes do not involve a significant reduction in the margin of safety.

In summary, based upon the above evaluation, I&M has concluded that these changes involve no significant hazards consideration.

ATTACHMENT 3 TO C1000-11

ENVIRONMENTAL ASSESSMENT

Indiana Michigan Power Company (I&M) has evaluated this license amendment request (LAR) against the criteria for identification of licensing and regulatory actions requiring environmental assessment in accordance with 10 CFR 51.21. I&M has determined that this LAR meets the criteria for a categorical exclusion set forth in 10 CFR 51.22(c)(9). This determination is based on the fact that this change is being proposed as an amendment to a license issued pursuant to 10 CFR 50 that changes a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR 20, or that changes an inspection or a surveillance requirement, and the amendment meets the following specific criteria.

(i) The amendment involves no significant hazards consideration.

As demonstrated in Attachment 2, this proposed amendment does not involve significant hazards consideration.

(ii) There is no significant change in the types or significant increase in the amounts of any effluent that may be released offsite.

The change in methodology would be applied to supplemental analyses for the determination of the time available for operator actions to prevent overfilling the secondary side of the affected steam generator in response to a steam generator tube rupture (SGTR) event. The present methodology would be retained for calculating the radiological consequences of a postulated SGTR since the current Donald C. Cook Nuclear Plant licensing basis methodology continues to bound the radiological consequences calculated by the new methodology. Therefore, there is no significant change in the types or significant increase in the amounts of any effluents released offsite.

(iii) There is no significant increase in individual or cumulative occupational radiation exposure.

The proposed change will not result in significant changes in the operation or configuration of the facility. There will be no change in the level of controls or methodology used for processing of radioactive effluents or handling of solid radioactive waste, nor will the proposal result in any change in the normal radiation levels within the plant. Therefore, there will be no significant increase in individual or cumulative occupational radiation exposure resulting from this change.

ATTACHMENT 4 TO C1000-11

COMMITMENTS

The following table identifies those actions committed to by Indiana Michigan Power Company (I&M) in this submittal. Other actions discussed in the submittal represent intended or planned actions by I&M. They are described to the Nuclear Regulatory Commission (NRC) for the NRC's information and are not regulatory commitments.

Commitment	Date
Revisions to the appropriate Unit 1 emergency operating procedures to incorporate enhancements identified due to the new steam generator tube rupture overfill methodology will be completed prior to restart from the current outage.	Prior to initial entry into Mode 2 from the current Unit 1 outage.
After NRC approval of the amendment request, the Updated Final Safety Analysis Report (UFSAR) will be revised to incorporate the use of the new methodology in steam generator tube rupture analyses.	To be included in the next 10 CFR 50.71(e) UFSAR update submittal.

Attachment 3

DC Cook response to Request for Additional Information Response

Indiana Michigan
Power Company
500 Circle Drive
Buchanan, MI 49107 1373

INDIANA MICHIGAN POWER

June 29, 2001

C0601-21
10 CFR 50.90

Docket Nos.: 50-315
50-316

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Mail Stop O-P1-17
Washington, DC 20555-0001

Donald C. Cook Nuclear Plant Units 1 and 2
RESPONSE TO NUCLEAR REGULATORY COMMISSION
REQUEST FOR ADDITIONAL INFORMATION
REGARDING LICENSE AMENDMENT FOR
"CHANGES IN STEAM GENERATOR TUBE RUPTURE ANALYSIS METHODOLOGY"
(TAC NOS. MB0739 AND MB0740)

References:
1) Letter from R. P. Powers (I&M) to Nuclear Regulatory Commission (NRC) Document Control Desk, "License Amendment for Changes in Steam Generator Tube Rupture Analysis Methodology," C1000-11, dated October 24, 2000.
2) Letter from J. F. Stang (NRC) to R. P. Powers (I&M), "Donald C. Cook Nuclear Plant, Units 1 and 2 – Request for Additional Information Regarding License Amendment Request," dated May 7, 2001, (TAC Nos. MB0739 and MB0740).
3) Letter from S. A. Greenlee (I&M) to NRC Document Control Desk, "Notification of a Due Date Extension for Response to a Request for Additional Information," C0501-18, dated May 31, 2001, (TAC NOS. MB0739 AND MB0740).

In Reference 1, Indiana Michigan Power Company (I&M), the Licensee for Donald C. Cook Nuclear Plant (CNP) Unit 1 and Unit 2, proposed to amend

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Facility Operating Licenses DPR-58 and DPR-74 to change the CNP licensing basis as described in the Updated Final Safety Analysis Report. I&M proposed to incorporate a supplemental methodology into its analysis of steam generator (SG) overfill following a postulated steam generator tube rupture (SGTR). The proposed change would use the analysis methodology documented in the Westinghouse Electric Company WCAP-10698-P-A, "SGTR Analysis Methodology to Determine Margin to Steam Generator Overfill," to more accurately calculate the transient response of CNP to a postulated SGTR with respect to SG overfill.

In Reference 2, the NRC informed I&M that additional information was needed to enable the NRC staff to adequately evaluate the proposed amendment. In Reference 3, I&M informed the NRC that the response to the requested information would be provided by June 30, 2001. The attachment to this letter provides the information requested in Reference 2.

There are no new commitments made in this submittal.

Should you have any questions, please contact Mr. Ronald W. Gaston, Manager of Regulatory Affairs, at (616) 697-5020.

Sincerely,

Scott A. Greenlee FOR

M. W. Rencheck
Vice President Nuclear Engineering

/bjb

Attachment

c: J. E. Dyer
MDEQ – DW & RPD
NRC Resident Inspector
R. Whale

ATTACHMENT TO C0601-21

RESPONSE TO NUCLEAR REGULATORY COMMISSION
STAFF REQUEST FOR ADDITIONAL INFORMATION
REGARDING LICENSE AMENDMENT FOR CHANGES IN STEAM GENERATOR TUBE
RUPTURE ANALYSIS METHODOLOGY

By letter dated October 24, 2000, from R. P. Powers, Indiana Michigan Power (I&M), to the Nuclear Regulatory Commission (NRC) Document Control Desk, I&M submitted a license amendment request to incorporate a supplemental methodology into its analysis of steam generator (SG) overfill following a postulated steam generator tube rupture (SGTR). The NRC, by letter dated May 7, 2001, from J. F. Stang (NRC) to R. P. Powers (I&M), requested additional information regarding the October 24, 2000, submittal. The information provided below responds to the NRC's request for additional information.

NRC Question

"Section D, 'Plant Specific Submittal Requirement,' of Enclosure 1 of the safety evaluation for WCAP-10698-P-A, states that certain plant-specific input shall be provided when referencing the WCAP for licensing actions. Please provide the required information."

I&M Response

I&M is proposing to utilize limited aspects of the WCAP-10698-P-A methodology and associated computer codes to supplement Donald C. Cook Nuclear Plant's (CNP's) current licensing basis SGTR analysis to directly address postulated steam generator overfill. The proposed change in methodology would be applied only to supplemental analyses for the determination of the time available for operator actions to prevent overfilling the secondary side of the affected SG in response to a SGTR event. This change in analysis technique is being proposed because it provides a more accurate representation of SG fill than CNP's current licensing basis. CNP's present methodology would be retained for calculating the radiological consequences of the postulated SGTR since the current CNP licensing basis methodology continues to bound the radiological consequences calculated by the new methodology. Since not all aspects of the WCAP-10698-P-A methodology are being adopted by I&M, not all five plant-specific input requirements, as described in Section D of the NRC safety evaluation report (SER) for WCAP-10698-P-A, are applicable to CNP.

The five items described in Section D of the SER for WCAP-10698-P-A are provided below with I&M's response following each item.

(1) **Each utility in the SGTR subgroup must confirm that they have in place simulators and training programs which provide the required assurance that the necessary actions and times can be taken consistent with those assumed for the WCAP-10698 design basis analysis. Demonstration runs should be performed to show that the**

accident can be mitigated within a period of time compatible with overfill prevention, using design basis assumptions regarding available equipment, and to demonstrate that the operator action times assumed in the analysis are realistic.

The required operator action times used in the CNP-specific SGTR margin to overfill analysis were verified via simulator demonstration runs to assure the operators could mitigate the accident within a period of time compatible with overfill prevention. The bounding design basis analysis documented in WCAP-10698 is not being used by CNP, as discussed further in item (5), since the margin to SG overfill following a SGTR has been addressed by CNP-specific analyses.

The required operator action times used in the CNP-specific analysis have been incorporated into CNP's Operator Training program to assure continued adequate performance and continued justification for the use of the assumed times. SGTR scenarios were also included in the simulator validation of CNP's Emergency Operating Procedure (EOP) program. Validation includes, for example, verification that the EOP is compatible with plant design and can be performed within acceptable time limits.

(2) **A site-specific SGTR radiation offsite consequence analysis which assumes the most severe failure identified in WCAP-10698, Supplement 1. The analysis should be performed using the methodology in SRP Section 15.6.3, as supplemented by the guidance in Reference (1).**

As stated in I&M's October 24, 2000, submittal, CNP's current licensing basis calculation for offsite radiological consequences of a postulated SGTR remains bounding. Specifically, the mass transfer values used in the current licensing basis analysis for offsite dose consequences bound those calculated using the new WCAP methodology. Therefore, CNP's present methodology is being retained for calculating the offsite radiological consequences of a postulated SGTR.

(3) **An evaluation of the structural adequacy of the main steam lines and associated supports under water-filled conditions as a result of SGTR overfill.**

An evaluation of the structural adequacy of CNP's main steam lines under water filled conditions is not necessary. The margin to SG overfill analysis following a SGTR determined that an overfilled condition would not occur. This conclusion is supported by CNP operator action timing and plant simulator runs, which have also been validated as discussed in item (1) above.

(4) **A list of systems, components and instruments which are credited for accident mitigation in the plant specific SGTR EOP(s). Specify whether each system and component specified is safety grade. For primary and secondary PORVs and control valves specify the valve motive power and state whether the motive power and valve controls are safety grade. For non-safety grade systems and components**

state whether safety grade backups are available which can be expected to function or provide the desired information within a time period compatible with prevention of SGTR overfill or justify that non-safety grade components can be utilized for the design basis event. Provide a list of all radiation monitors that could be utilized for identification of the accident and the ruptured steam generator and specify the quality and reliability of this instrumentation if possible. If the EOPs specify steam generator sampling as a means of ruptured SG identification, provide the effect on the duration of the accident.

As indicated in CNP's response to item (1), the license amendment request is intended to apply a more accurate analysis to demonstrate that CNP's current licensing basis of precluding SG overfill is maintained. A list of the systems, components and instruments credited in CNP's current SGTR analysis is provided in the table below.

The actions prescribed in CNP's EOPs for SGTR accident mitigation focus on four key aspects:

- isolating the ruptured SG,
- cooling the reactor coolant system (RCS),
- depressurizing the RCS, and
- terminating operation of the emergency core cooling system (ECCS).

The EOPs for a reactor trip or safety injection, and a steam generator tube rupture direct the operators to use specific equipment. The most challenging SGTR scenario with respect to SG fill includes a coincident loss of offsite power, which results in some of the equipment allowed to be used in the EOPs (consistent with the Westinghouse Emergency Response Guidelines) to be unavailable. Thus, the EOP-prescribed equipment presented in the tables and discussions below are based on offsite power not being available.

Equipment/ Component Name	ID No.	Safety Grade (Y or N)	If Non-Safety Grade, is Safety Grade Backup Available? (Y or N)	Function	Remarks
SG Water Level – Narrow Range	BLP-1x0, -1x1, -1x2; where x corresponds to SG's 1 through 4	Y	N/A	Ruptured SG Identification	
AFW Discharge Valve	FMO-211 & -212, -221 & -222, -231 & -232, or -241 & -242	Y	N/A	Ruptured SG Isolation	
Main Feed Pump Trip	Unit 1: 1-OME-84-LPSVE, -LPSVW, -HPSVE, -HPSVW Unit 2: 2-OME-84-SCVE, -SCVW	N	N	Feedwater Isolation	Notes 1 and 6

Main Feed Pump Discharge Valves	FMO-251 and –252	N	N	Feedwater Isolation	Notes 1 and 6
Feedwater Regulating Valves	FRV-210, -220, -230, -240	N	N	Feedwater Isolation	Notes 1 and 6
Feedwater Isolation Valves	FMO-201, -202, -203, -204	N	N	Feedwater Isolation	Notes 1 and 6
Steam Supply Valve to TDAFP	MCM-221, or –231	Y	N/A	Ruptured SG Isolation	
SG Blowdown Isolation Valve	DCR-310, -320, -330, or –340	N	N	Ruptured SG Isolation	Note 6
SG Blowdown Sample Valve	DCR-301, -302, -303, or –304	N	N	Ruptured SG Isolation	Note 6
SG Stop Valve	MRV-210, -220, -230, or –240	Y	N/A	Ruptured SG Isolation	Note 2
SG PORVs (air-operated)	MRV-213, -223, -233, -243	N	N	Ruptured SG Isolation and RCS Cooldown via Non-Ruptured SGs	Notes 3, 4, and 6
Core Exit Thermocouples	NTR-1 through –65	Y	N/A	Monitor RCS Cooldown	
Pressurizer PORVs (air-operated)	NRV-151, -152, and –153	N	N	RCS Depressurization	Notes 4 and 6
Pressurizer Water Level	NLI-151, NLP-151, -152, -153	Y	N/A	ECCS Termination Criteria	Note 5

<u>Notes:</u>

1. The isolation of main feedwater is provided post-reactor trip via a feed pump trip, closure of the feed pump discharge valves (FMO-251 & -252), feed regulating valves (FRV-210, -220, 230, and -240), and the feedwater isolation valves (FMO-201, -202, -203, and –204).

2. The SG stop valves drain valve (DRV-407) and steam line warming valves (MS-144 and -143) are also closed, or check closed. These lines are not a significant secondary vent path, and thus timely closure is not critical.

3. The SGTR EOP directs the operators to increase the setpoint of the SG PORV on the ruptured SG from the nominal setpoint of 1025 psig to 1040 psig.

4. The SG PORVs form part of the main steam system pressure boundary upstream of the SG stop valves, and thus are safety-grade. However, the electrical and control air appurtenances for the SG PORVs are non-safety grade. Similarly, the pressurizer PORVs provide part of the safety grade RCS pressure boundary, but the control functions to manipulate these valves are classified as non-safety grade. Two of the

three pressurizer PORVs (NRV-152 and –153) have a continuous back-up source of air in the form of air bottles inside containment. As such, no operator action is required to align the back-up air bottles.

5. During the RCS depressurization, pressurizer level is restored to satisfy, in part, the ECCS termination criteria. Upon satisfying the ECCS termination criteria, the safety injection pumps are secured, all but one centrifugal charging pump (CCP) is stopped, and the boron injection tank is isolated. The remaining CCP is used to establish charging and letdown.

6. The Unit 1 and Unit 2 CNP licensing basis SGTR analysis described in Section 14.2.4 of the UFSAR credits non-safety grade equipment to mitigate the consequences of a SGTR accident. Thus, the use of the non-safety grade equipment, (e.g., SG and pressurizer PORVs) for accident mitigation following a SGTR in the margin to SG overfill analyses is consistent with CNP's licensing bases for both units.

The following radiation monitors could be used for identification of a SGTR accident.

- SG blowdown line,
- SG power-operated relief valve (PORV) line,
- gland steam condenser vent, and
- steam jet air ejector vent.

All four of the radiation monitor instruments listed above are non-safety grade, and are specifically listed in the EOPs as indications that can be used for identification of a SGTR event. These instruments are maintained and tested in accordance with CNP Technical Specifications and Offsite Dose Calculation Manual requirements. In accordance with 10 CFR 50.65, these instruments are also monitored within the scope of CNP's maintenance rule program. These monitors have not exceeded their maintenance reliability criteria.

The SGTR EOP allows identification of the ruptured SG by any of following indications:

a) directing chemistry personnel to sample all SGs for an activity analysis and perform a one-minute beta analysis, and

b) identifying the ruptured SG by one or more of the three indications:
   - an unexpected rise in SG narrow range level, or
   - high radiation from any SG sample, or
   - high radiation from any SG PORV.

The most limiting accident scenario with respect to SG tube overfill is the complete severance of one SG U-tube. The high primary to secondary flows that result from this design basis accident allow the ruptured SG to be promptly identified by the operators due to the unexpected rise in SG narrow range level. Thus, the most timely indication of the design basis SGTR is provided by the narrow range SG water level.

The chemistry sample provides a longer-term confirmatory indication of the ruptured SG for large break scenarios. The chemistry sample is only one of three methods specified in the SGTR EOP for the identification of the affected SG. The operator can use any one of the three methods for identifying a ruptured SG. The EOP actions focus primarily on the other two methods; checking the readings from the four radiation monitors discussed previously to see if the SG tubes are intact, and checking SG levels. The reactor trip EOP does include directions to sample the SGs in the event that the ruptured SG event was not identified by checking the radiation monitors. Thus, the chemistry sample of a ruptured SG provides confirmation, not primary indication, in the case of a design basis break.

In summary, the high primary to secondary flow rate due to a design basis break results in SG water level behavior and radiation monitor alarms that provide relatively quick indication of a ruptured SG. Therefore, the SG chemistry sample analysis time is not a critical aspect of the accident mitigating actions for scenarios that would be challenging with respect to SG fill.

(5) **A survey of plant primary and "balance-of-plant" system design to determine the compatibility with the bounding plant analysis in WCAP-10698. Major design differences should be noted. The worst single failure should be identified if different from the WCAP-10698 analysis and the effect of the difference on the margin of overfill should be provided.**

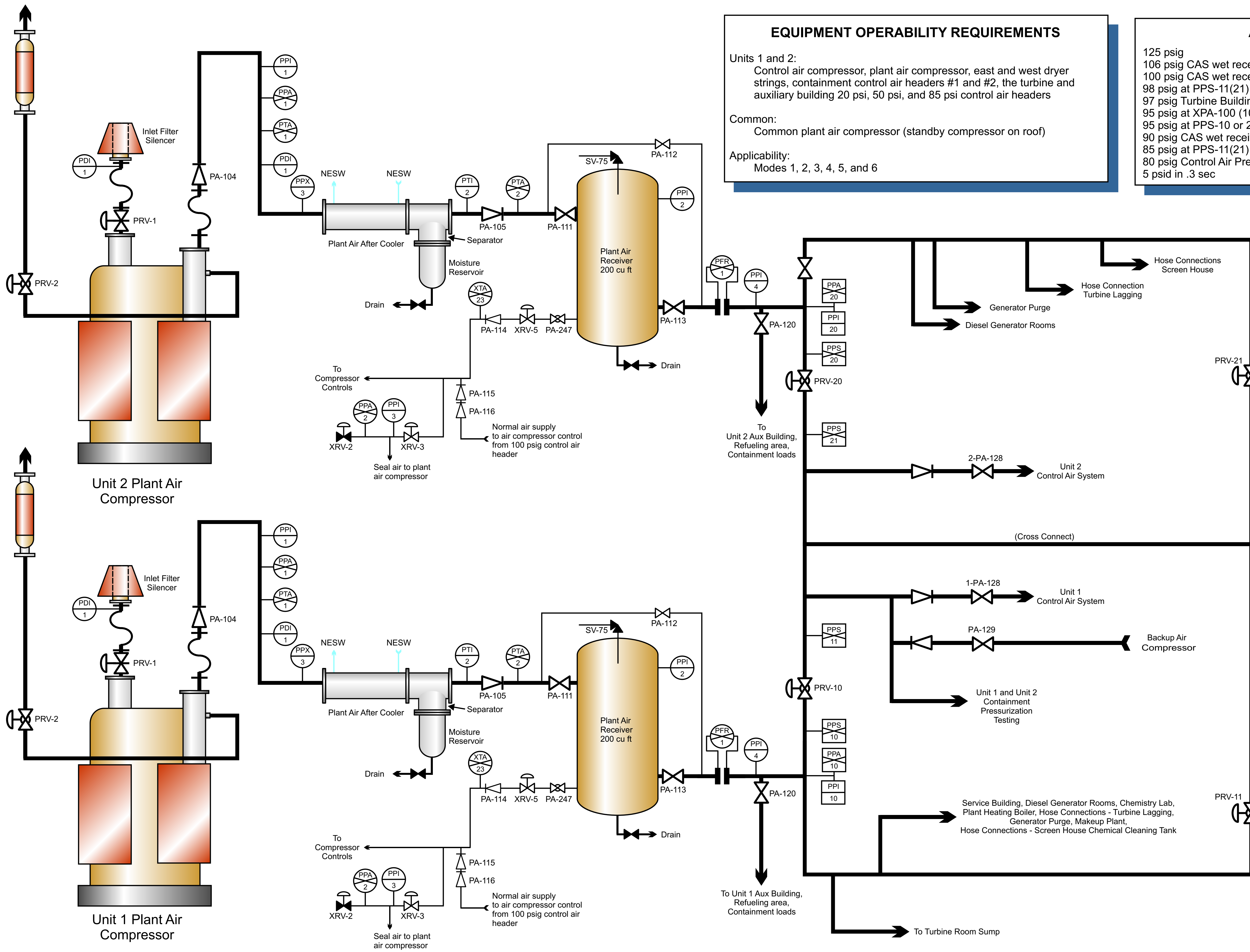
This item is not applicable to CNP. CNP is not using the bounding plant analysis as described in WCAP-10698. As stated in Reference 1, CNP proposes to apply limited use of the WCAP-10698 methodology (and associated computer code) to supplement the CNP SGTR analysis with a CNP-specific margin to SG overfill analysis. The supplemental margin to SG overfill analysis used plant-specific analysis input values. Since the supplemental analysis is CNP-specific, the WCAP-10698 bounding plant analysis was not used, or needed. Therefore, there is no need to differentiate the CNP design from the generic plant design. Regarding worst single failure assumptions, CNP's current licensing basis SGTR analysis does not include a single failure. Thus, there is no need for CNP to consider a single failure in the SGTR margin to overfill analysis.

Comparison to No Significant Hazards Evaluation

CNP's response to the NRC's request for additional information does not affect the original evaluation performed in accordance with 10 CFR 59.92. The information provided in this response is considered supporting information and does not change the intent of CNP's original submittal.

Attachment 4

Compressed Air and Control Air System Overview Drawings



EQUIPMENT OPERABILITY REQUIREMENTS

Units 1 and 2:
Control air compressor, plant air compressor, east and west dryer strings, containment control air headers #1 and #2, the turbine and auxiliary building 20 psi, 50 psi, and 85 psi control air headers

Common:
Common plant air compressor (standby compressor on roof)

Applicability:
Modes 1, 2, 3, 4, 5, and 6

AIR SYSTEM SETPOINTS

125 psig	Air receiver safety valves open
106 psig CAS wet receiver pressure	CAC unloads
100 psig CAS wet receiver pressure	CAC loads
98 psig at PPS-11(21)	Plant air header un-isolates
97 psig Turbine Building air header	Alarm "PAC failure / low pressure"
95 psig at XPA-100 (100# header)	Control air pressure low alarm
95 psig at PPS-10 or 20	Stand-by PAC starts
90 psig CAS wet receiver pressure	Associated CAC auto-starts
85 psig at PPS-11(21)	Plant air header isolates
80 psig Control Air Pressure	Manual Reactor Trip
5 psid in .3 sec	PAC surge protection-unloader opens

SYSTEM PURPOSE

The Plant Air System provides compressed air for general plant services, instrumentation and control (the Plant Air System normally supplies the Control Air System), containment integrated leak rate testing, containment penetration pressurization, and respiratory use.

AUTO FEATURES

- Compressor auto starts at 95 psig in the plant air ring header by PPS-10(20)
- Plant air ring header auto isolation by PPS-11(21)
- Compressor unloader valve PRV-2 maintains constant discharge pressure with changing demand
- Compressor unloader valve fully opens for surge protection
- Auxiliary oil pump auto start on PAC Start and SHuddown

PAC AUTO TRIPS

- Load Shed (Train B)
- Load Conservation (Train B)
- Low Oil Pressure
- High Oil Temperature
- High 1st Interstage Air Temperature
- High 2nd Interstage Air Temperature
- Loss of Control Inverter
- Microprocessor Fault
- Motor Current Transducer Failure
- Transmitter Failures

PAC MANUAL TRIPS

- Control Room Switch
- DCCS- Select STOP
- Manual Emergency Trip PB

SOD-06401-002
PLANT AIR SYSTEM

Sheet #: 1	Rev. 5	Date: 09/28/10

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EQUIPMENT OPERABILITY REQUIREMENTS

Units 1 and 2:
Control air compressor, plant air compressor, east and west dryer strings, containment control air headers #1 and #2, the turbine and auxiliary building 20 psi, 50 psi, and 85 psi control air headers

Common:
Common plant air compressor (standby compressor on roof)

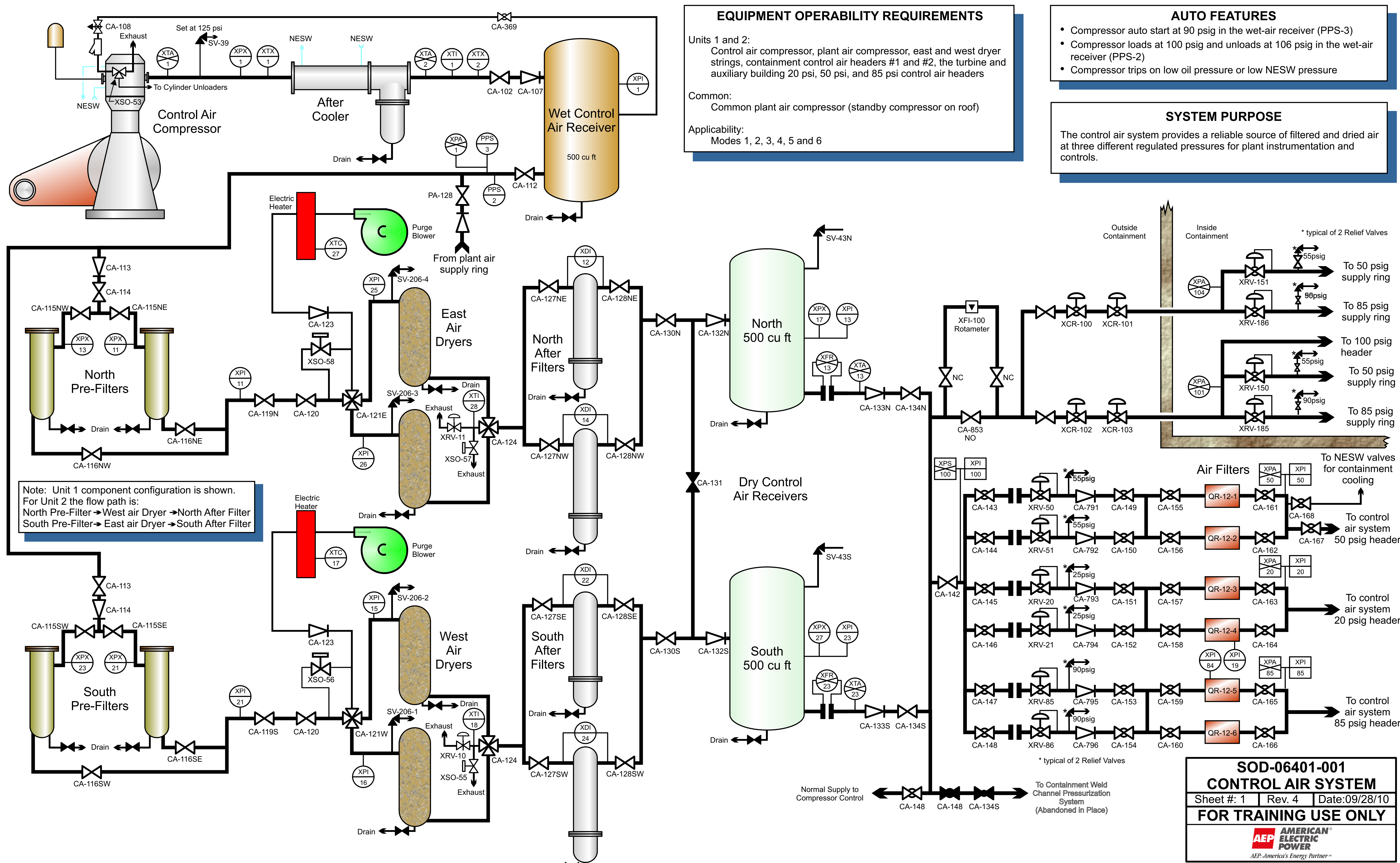
Applicability:
Modes 1, 2, 3, 4, 5 and 6

AUTO FEATURES

- Compressor auto start at 90 psig in the wet-air receiver (PPS-3)
- Compressor loads at 100 psig and unloads at 106 psig in the wet-air receiver (PPS-2)
- Compressor trips on low oil pressure or low NESW pressure

SYSTEM PURPOSE

The control air system provides a reliable source of filtered and dried air at three different regulated pressures for plant instrumentation and controls.



SOD-06401-001
CONTROL AIR SYSTEM

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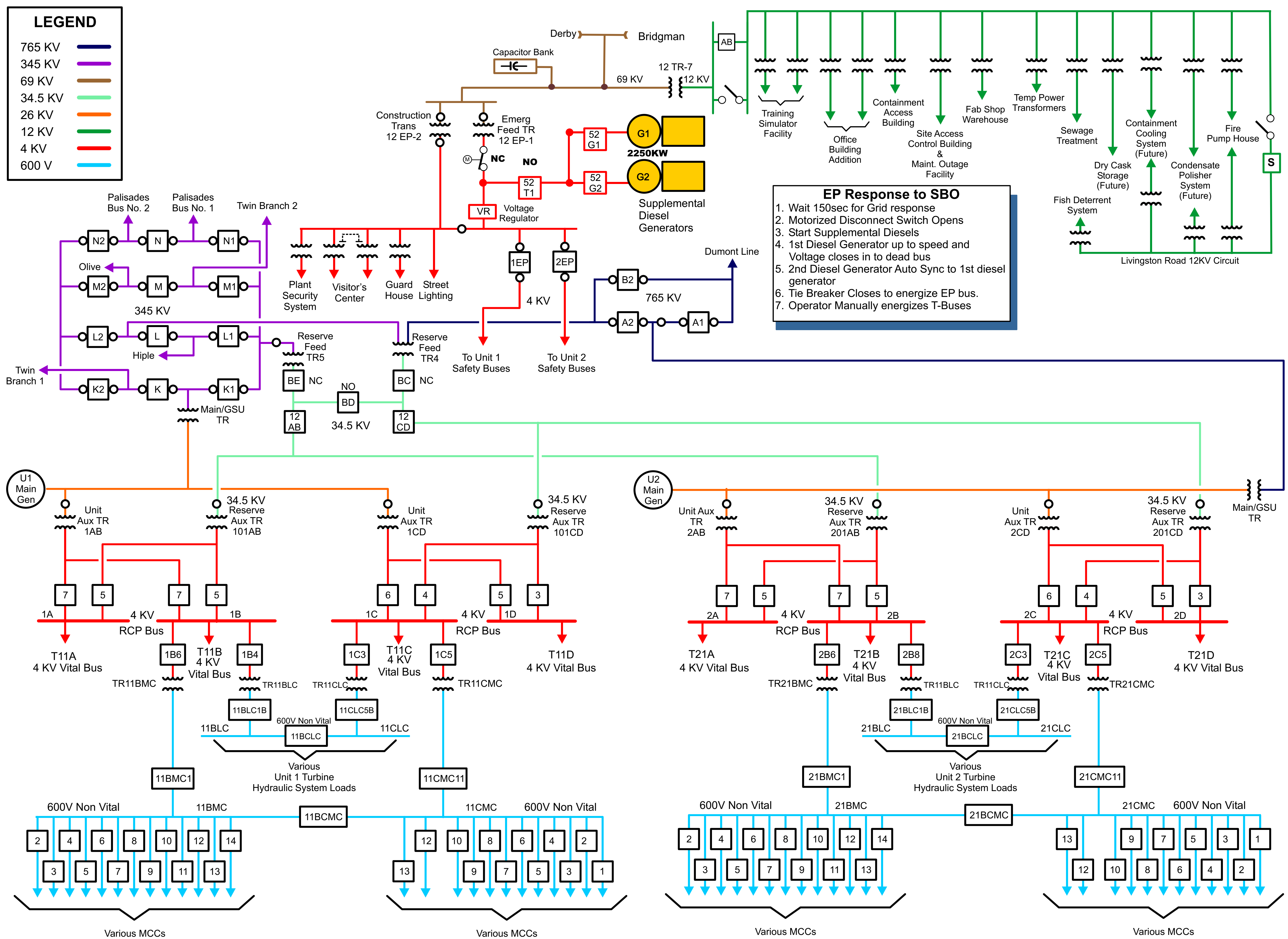
Attachment 5

Electrical System Overview Drawings



PURPOSE

The BOP Electrical System has no Safety Related functions, and no credit is taken for the offsite power sources for Safe Shutdown due to Appendix R Fire or Station Blackout. Non-Safety Related Functions are:

- Connect the main generator output to the AEP System grid.
- Supply power to all plant electrical buses including the Reactor Coolant Pump (RCP) buses, 600 VAC Buses BMC and CMC, and smaller MCCs served by BMC and CMC. The BOP System uses an offsite power source when the plant is off-line.
- Supply power for support buildings and functions.

RCP BUS FAST TRANSFER

Fast Transfer is the automatic re-alignment of power supplied to the RCP buses from the UATs, to the RATs as follows:

- Trip the generator output switchyard circuit breakers.
- Trip the UAT supply breakers to the RCP buses.
- Close the RAT supply breakers to the RCP buses.

These actions initiate simultaneously resulting in 30 milliseconds dead time on the RCP buses. Fast transfer is designed to maintain power to the RCP motors.

For lockout relay activated generator trips, the RCP buses immediately fast transfer. For reactor / turbine trips, there is a 30 second delay while the generator remains connected to the electrical system, then fast transfer initiates.

SOD-08200-001 BALANCE OF PLANT ELECTRICAL SYSTEM

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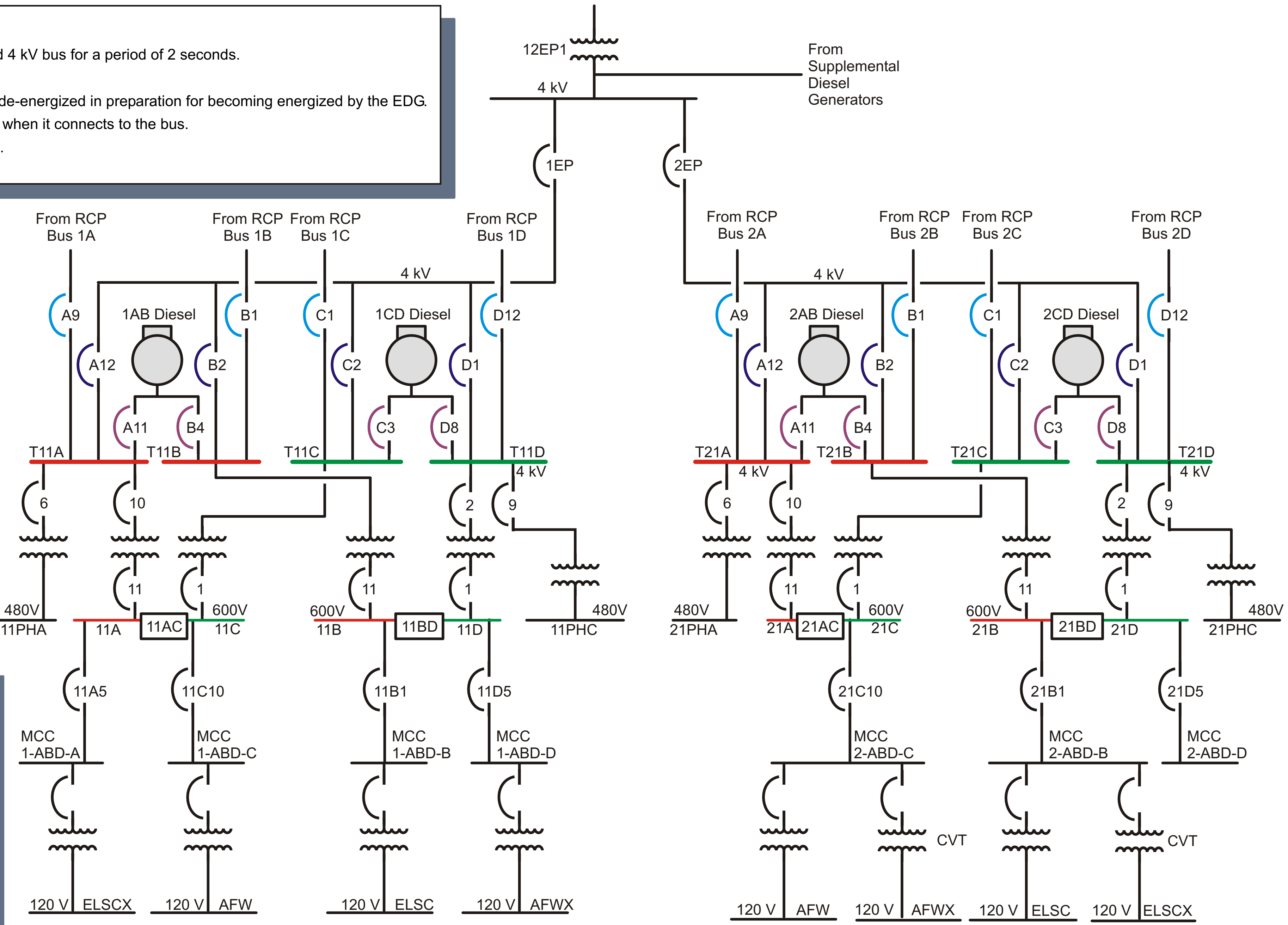
LOAD SHED

A load shed signal is developed in response to an undervoltage (80% of nominal) sensed on any 2 of 3 phases of any safety related 4 kV bus for a period of 2 seconds.

Actions that occur upon receipt of a load shed signal:

1. Trips open and blocks closure of the feed breakers which supply power to the bus, thereby ensuring the bus is completely de-energized in preparation for becoming energized by the EDG.
2. Trips open and blocks closure of breakers that supply power to the large motor loads to prevent the EDG from overloading when it connects to the bus.
3. Trips open and locks out selected small loads to prevent overloading the EDG as ESF equipment sequences on to the bus.
4. Starts the EDG.



TECHNICAL SPECIFICATIONS

3.3.5 Loss of Power (LOP) Diesel Generator (DG) Start Instrumentation
3.8.1 AC Sources - Operating
3.8.2 AC Sources - Shutdown
3.8.9 Distribution Systems - Operating
3.8.10 Distribution Systems - Shutdown

BREAKER INTERLOCKS

69 kV emergency feeder breakers to 4 kV vital buses

- Emergency feeder breakers will not close if either EDG or RCP bus tie breaker is closed
- EDG & RCP bus tie breakers will not close if 69 kV Breaker is closed

Purpose - Prevent paralleling emergency feeder with reserve & auxiliary power supplies (emergency busses 30° out of phase with offsite)

RCP bus tie breakers

- Cannot close RCP bus tie breakers (A9, D12, etc.) unless either reserve or auxiliary feeder breakers to the RCP bus is closed

Purpose - Prevent powering RCPs from EDGs

11B & 11D bus tie breaker (same for 11A & C tie)

- Bus tie breaker cannot be closed unless feeder breaker for either 11B or 11D is open

Example: Can't close 11BD unless either 11B11 or 11D1 is open

Purpose -Prevent inadvert paralleling of EDGs or opposite electric trains which could cause overcurrent condition.

Note - These breakers are Red Tagged open.

11PHA & C feed to variable control group PZR heaters

- Only one at a time can be closed

Purpose - Prevent paralleling EDGs on blackout

Normal at power all RCP buses from our main generator output

LOAD CONSERVATION

The load conservation circuit strips non-essential loads from the safeguards buses in the event of a LOOP concurrent with a SI or CTS signal.

Conditions that cause load conservation:

1. LOOP followed by a SI - EDG up to speed, output breakers closed to T11A/21A or D, and SI occurs.
2. SI followed by a load shed - Standing SI signal, EDG up to speed but not connected to the bus, and a bus undervoltage condition occurs.
3. LOOP followed by a CTS signal - EDG up to speed, output breakers closed to T11A/21A or D, and a CTS signal occurs.
4. CTS followed by a load shed - Standing CTS signal, EDG up to speed, and a bus undervoltage condition occurs.

Actions that occur upon receipt of load conservation signal:

1. Trips and locks out selected non-essential vital bus loads.
2. Prevents restarting any stripped loads until either the 75 second load conservation timer times out or SI is reset after the 60 second SI timer times out.
3. Trips non-essential service water pump.

If another SI signal is generated after the SI is reset with the EDG breakers closed, load conservation reinitiates.

DEGRADED BUS VOLTAGE

Degraded Bus Voltage Sensed
(2/3 Phase on T11A(T21A) or T11D(T21D) at 113V Indicated)

Without SG Lo-Lo Level OR SI Signal AND on UAT's

- After 90 seconds the train sensing the DBV will transfer to the RAT.
- The RAT will load change to try to address the low voltage condition.
- IF the low voltage condition persists for 21 seconds, the train RCP Tie Breakers Open.
- Note: 9 seconds to sense + 90 second wait for swap to RAT + 21 seconds to correct = 120 seconds.

With SG Lo-Lo Level OR SI Signal

- After **9 seconds**, RCP Tie Breakers open.
- Loss of voltage relays Start the EDGs

BUS T11A	BUS T11D
1S SI	1N SI
West AFW Pump	East AFW
1W CS Pump	1E CS Pump
1W RH Pump	1E RH Pump
1W ESW Pump	1E ESW
1W CCW	1E CCW
1W Charging Pump	1E Charging Pump

LEGEND

Train A
Train B
RCP bus tie breaker
Emergency feeder breaker
EDG feeder breaker

EMERGENCY FEED TRANSFORMER (12 EPI)

Used only if
- Normal power lost &
- Reserve power lost &
- One or both DGs lost

SOD-08201-001
ENGINEERED SAFETY SYSTEMS ELECTRICAL

Sheet #: 1	Rev. 4	Date: 03/09/09

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Attachment 6

DC Cook UFSAR Section 1.3.9.h

1.3.8 SAFETY FEATURES

The engineered safety features provided for this plant have sufficient redundancy of components and power sources such that under the conditions of a loss-of-coolant accident they can maintain the integrity of the containment and keep the exposure of the public below the limits of 10 CFR 100, even when operating with partial effectiveness. The safety features incorporated in the design of this plant and the functions they serve are summarized below.

a) The Emergency Core Cooling System (ECCS) injects borated water into the Reactor Coolant System. The ECCS limits damage to the core and limits the energy and fission products released into the containment following a loss-of-coolant accident.

b) A steel-lined, domed, reinforced concrete containment vessel is anchored to a reinforced concrete foundation slab. The containment is designed to remain virtually leaktight during the pressure transient following a loss-of-coolant accident.

c) An Ice Condenser System reduces containment pressure and removes iodine radioactivity following a loss-of-coolant accident.

d) A Containment Spray System is used to reduce containment pressure and to remove iodine from the containment atmosphere following a loss-of-coolant accident.

e) The Containment Isolation System incorporates valves and controls on piping systems penetrating the containment structure. The valves are arranged to provide two barriers between the Reactor Coolant System or containment atmosphere and the environment. System design is such that failure of one valve to close will not prevent isolation, and no manual operation is required for immediate isolation. Automatic Phase "A" isolation is initiated by a containment isolation signal derived from the safety injection automatic activation logic and Phase "B" isolation from a containment pressure high-high signal.

f) Reliable on-site diesel-generator power is provided for the engineered safeguards loads in the event of failure of station auxiliary power. In addition, even if external auxiliary power to the station is lost concurrent with an accident, power is available for the engineered safeguards from on-site diesel-generator power to assure protection of the public health and safety for any loss-of-coolant accident.

g) The active components necessary for the proper operation of the engineered safety features are operable from the control room.

The Engineered Safety Features in this plant are the ECCS, the containment structure, the Ice Condenser System, and the Containment Spray System (items a, b, c, d above).

1.3.9 SHARED FACILITIES AND EQUIPMENT

Separate and similar systems and equipment are provided for each unit except as noted below. In those instances where components of a system are shared by both units, those components, which are shared, are either shown in the following listing or discussed in the applicable Sub-Chapter.

1.3.9.a Chemical And Volume Control System

Item	Number Shared
Boric Acid Tanks	3
Batching Tank	1
Hold-up Tanks	3
Boric Acid Reserve Tank	1
Recirculation Pump	1
Boric Acid Evaporator Feed Pumps	3
Evaporator Feed Ion Exchangers	4
Boric Acid Evaporator	2
(Converted to a radioactive waste evaporator) (See Section 11.1)	
Monitor Tanks	4
Monitor Tank Pumps	2
Evaporator Condensate Demineralizers	2

1.3.9.b Spent Fuel Pit Cooling System

Item	Number Shared
Spent Fuel Pool Pumps	2
Spent Fuel Pool Demineralizer	1
Spent Fuel Pool Filter	1
Spent Fuel Pool Heat Exchangers	2
Refueling Water Purification Pump	1

1.3.9.c Fuel Handling System

Item	Number Shared
Spent Fuel Storage Pool	1
New Fuel Storage Area	1
Decontamination Area	1
Spent Fuel Pool Bridge Crane	1

1.3.9.d Service Water Systems

Item	Number Shared
Essential Service Water Pumps	4

Item	Number Shared
Non-Essential Service Water Pumps	4

1.3.9.e Auxiliary Steam System

1.3.9.f Waste Disposal System

1.3.9.g Radiation Monitoring System

1.3.9.h Structures, Buildings And Miscellaneous

Item
Auxiliary Building
Fuel Handling Area
Service Building
Lake Intake Structures
Compressed Air Services
Plant Heating Steam System
Make-up Water Supply and Treatment System
Non-Essential Service Water System
Seismic Monitoring System
Post Accident Sampling System

1.3.9.i Component Cooling Water System

Item	Number Shared
Component Cooling Water Pumps	1

1.4 PLANT SPECIFIC DESIGN CRITERIA (PSDC)

The criteria followed in the design of the Donald C. Cook Nuclear Plant have been developed as performance criteria which define or describe safety objectives and procedures, and they provide a guide to the type of plant design information which is included in this UFSAR. These plant specific design criteria define the principal criteria and safety objectives for the design of the Cook Plant. A complete set of these criteria is stated explicitly in the following Sections. The safety objectives and procedures are then more fully described in other sections of the UFSAR.

The Atomic Energy Commission (AEC) published proposed GDCs for public comment in 1967. The Atomic Industrial Forum (AIF) reviewed these proposed criteria and recommended changes. The Cook plant was designed and constructed to meet the intent of the Proposed General Design

Attachment 7

DC Cook UFSAR Section 9.8.2

Design Evaluation

Since all the fire pumps are separated from each other by construction and distance, as well as by the ability to isolate a given pump and its section of header in the event it is damaged, it is highly unlikely that the entire pump and piping system will ever be lost. The fire pumps are so sized that any two of them can supply the water demand for the largest single hazard anticipated.

The fire protection water system is designed so that the fire pumps, water storage tanks and associated equipment are common to both units. Any pump is capable of supplying water to any automatic system, regardless of unit, and either control room can automatically start all three primary pumps. However, it is also possible to completely isolate one unit from the other hydraulically and electrically and operate each independently. In this manner, the equipment from one unit can be used and support the other unit, but should severe damage occur to the fire system in one unit, the fire system for the other unit can be isolated and will continue to function.

Tests and Inspections

Fire protection/detection equipment is periodically tested in accordance with Technical Requirements Manual (TRM) to insure proper performance when required. In addition, appropriate surveillance and tests are performed on all portable equipment to insure that it is properly located, charged, and in good working condition.

Fire drills are held regularly to maintain the fire fighting capability of the plant fire brigade at a high level.

9.8.2 COMPRESSED AIR SYSTEM

The Compressed Air System is shown on Figure 9.8-3.

9.8.2.1 Design Bases

Parameters included in design:

1. The system must provide redundant compressed air supplies for control and instrument air requirements.
2. The system must provide adequate compressed air capacity for:
   a. General Plant Service
   b. Control
   c. Instrumentation
   d. Testing
   e. Containment Penetration and Weld Channel Pressurization System
   f. Respiratory protection in the containment structure itself, as per compressed gas association commodity Spec. G-7.1 - 1966, per OSHA Standards and Interpretations 1910.134.
3. The system must provide a continuous supply of compressed air to vital systems under both normal and abnormal conditions.

9.8.2.2 System Description

The Compressed Air System includes the combined service and control instrument air sub-systems, the air supply for the Containment Penetration and Weld Channel Pressurization System and air respiratory protection at strategic location. Either of the two full capacity plant air compressors, one located in the turbine building of each unit, is capable of supplying compressed air to the plant air receivers for general service air for both units. In addition, the plant air compressors supply air to the dry control-instrument air receivers through redundant pre- and after-filters and dryers.

The Containment Penetration and Weld Channel Pressurization System air receivers are supplied from these dry control-instrument air receivers.

In addition, a standby control air compressor capable of supplying the control and instrument air for its unit is installed as a backup for the normal control-instrument air supply, i.e., the plant air compressors. This standby compressor is also capable of supplying air to the Containment Penetration and Weld Channel Pressurization System for its unit. The standby control air compressor is designed to start automatically upon detection of low air pressure in the plant air header.

The four air compressors (two plant, two control) are of the oil-free type to eliminate oil contamination of the control instrument air. Control and instrument air is also filtered and dried to remove any particulate matter and/or moisture which could interfere with the operation of any instrumentation and control equipment.

The Control Air System includes sufficient capacity to supply the control and instrument air requirements with the equivalent of approximately 5 minutes of control air output after a loss of power incident. Additionally, certain vital control valves within the containment are each equipped with a local receiver tank with capacity to activate the valve. Also, the control air compressors can be supplied with electric power from both normal and emergency sources so that a supply of compressed air can be made available in any foreseeable circumstance.

The Compressed Air System includes normal accessory equipment such as dryers, filters, storage receivers, after-coolers, and safety valves in addition to the compressors. A descriptive summary of the major pieces of equipment in the system is included in Table 9.8-2.

In addition, a 650CFM backup plant air compressor is installed on the Unit # 1 4KV Switchgear Room roof. The backup plant air compressor is tied into the normal plant compressed air system at both the inlet and outlet of the Containment Test Pressurization Air Dryer and can supply any service air or control instrument air load. The compressor must be manually started and aligned. The 575 V, 150 Amp power supply is provided from 600V Bus 11CMC.

9.8.2.3 Design Evaluation

The Compressed Air System is designed to provide a reliable source of compressed air for all plant uses.

During normal operation, either one of the two plant air compressors is capable of supplying the entire demand of both plant and control-instrument air requirements for both units.

Low plant air header pressure will automatically start the second plant air compressor. The air compressor is then manually loaded and placed in automatic pressure control. A lower control air header pressure in either unit will automatically start that unit's control air compressor. A further degradation in the plant air header pressure will cause the four air-operated isolation valves located in the plant air ring header to close, thus completely isolating the control air systems of the two units.

This system arrangement allows either unit's plant air system to be removed from service should that become necessary while allowing the remainder of the plant air system as well as both unit's control air system to continue in operation. This isolation can be achieved by closing the two air-operated isolation valves, which serve the effected unit.

In this manner, each unit still retains a backup supply of compressed air from its own control air compressor.

A failure in the control air system of one unit will not affect the control air system of the other unit because check valves in the control air off-takes from the plant air header prevent back flow.

Each control air header has safety relief valves to protect against over-pressurization.

9.8.2.4 Tests and Inspections

The compressed air systems is in service during all modes of operation. Flow and pressure instrumentation for the plant air system, and the control air system permit monitoring the systems for excessive air consumption, or inadequate compressor capacity.

Tests and inspections include:

- Routine functional testing of the standby plant and control air compressors, to ensure their readiness.
- Routine compressor inspections /overhauls.
- Routine prefilter, dryer desiccant, and after filter inspection/replacement.
- Routine air quality monitoring for moisture, and breathing air quality.

9.8.3 SERVICE WATER SYSTEMS

The Service Water Systems are shared by both units.

9.8.3.1 Design Basis

The Service Water Systems supply cooling water to various heat exchangers in both the primary and secondary systems of each unit. Provisions are made to ensure a continuous flow of cooling water to those systems and components necessary for plant safety both during normal operation or under accident conditions. Sufficient redundancy of piping and components is provided to insure that cooling is maintained to vital services at all times.

Attachment 8

Applicable Pages of DC Cook Procedure PMP-2291-WMP-001

Doc No.:	PMP-2291-WMP-001	Rev No.: 020
Title:	WORK MANAGEMENT PROCESS FLOWCHART	

Alteration Cat.: Minor Revision
CDI/50.59: N/A

PORC Mtg. No.: N/A
CARB Mtg. No.: 675
Admin Hold AR No.: N/A
Superceding Proc(s): N/A

Temp Proc Exp Date: N/A
Temp Change Exp Date: N/A
Temp Proc/Change End: N/A
Effective Date: 8/24/2012 12:00:00 AM

Approvals

Name	Review/Approval Type/Capacity	Date
Wendzel, Regan	7 Approval Authority	08/15/2012 12:30
Maret, Bruce	5 Management Review	08/14/2012 10:31
Hicks, Roger	3 Technical Review	08/08/2012 13:41
Carlson, Guss E	1 Validation Review	08/08/2012 13:25
Lain, Thomas	3 Technical Review	07/20/2012 10:19
Keppeler, Randall	1 Cross-Discipline Review	07/20/2012 08:49
Lain, Thomas	1 Cross-Discipline Review	07/09/2012 12:04
Williams, Alisa	1 Cross-Discipline Review	07/06/2012 11:33
Carlson, Guss E	1 Validation Review	07/03/2012 11:56
Swanson, Scott	1 Cross-Discipline Review	06/29/2012 09:25
Gressley, Scott	1 Cross-Discipline Review	06/28/2012 22:17
Gressley, Scott	1 Cross-Discipline Review	06/18/2012 13:36

Signature Comments

Approved per Plant Manager Q. Shane Lies.
Approved at CARB 675
Will perform final review of BOM reviews with PM upon his return from vacation. Until then, this is approved as written.
Comments provided via e-mail

Inoperable or Unavailable Item	Equipment Guarded	Posting requirements	Basis
Charging Pump (CCP)	Operable CCP	Rope across access to pump cubicle, sign with flasher on pump and sign on rope. Sign on breaker and tag in Control Room at 101 switch.	Protection of operable Tech Spec equipment
Component Cooling Water (CCW) Pumps	Operable CCW Pump	Signs with flasher on pump. Sign on 4kV breaker and tag in Control Room at 101 switch.*	Protection of operable Tech Spec equipment. Loss of all CCW requires reactor trip and trip of all RCPs within 2 minutes.
Condensate Booster Pump	Both available Condensate Booster Pumps **AND** Running Heater Drain Pumps	Signs with flashers on pumps. Sign on 4kV breakers and tags in Control Room at each 101 switch.*	Redundant equipment and equipment to mitigate loss of additional Condensate Booster Pump
Containment Cooling Chiller Package	Redundant Containment Cooling Chiller Package	Signs with flashers on control panels for the individual Containment Cooling Chiller Package. Signs on breakers. Sign on the integrated control console.	Redundant equipment and potential impact on Containment Cooling capability.
Containment Spray	Operable CTS Train	Sign with flasher on pump room gate and on stairs leading to heat exchanger, tag at 101 switch and sign on pump breaker.	Protection of operable Tech Spec equipment
Control Air Compressor (CAC)	Both PACs	Sign with flasher on PAC. Sign on breaker and tag in Control Room at 101 switch.*	Redundant equipment
Control Air Dryer/Filter/ Regulator	Redundant equipment	Sign with flasher on component. Sign on breaker and control switches when applicable.	Redundant equipment

Inoperable or Unavailable Item	Equipment Guarded	Posting requirements	Basis
Motor Driven Auxiliary Feedwater Pump (MDAFP)	Other MDAFP **AND** TDAFP	MDAFW Pump – sign with flasher on door to East pump or sign with flasher on gate to West pump. Sign on 4kV breaker for applicable pump and tag in Control Room at applicable 101 switch. TDAFW Pump – sign with flasher on door to pump room and tag in Control Room at 101 switch.	Protection of operable Tech Spec equipment. Redundant equipment
Motor Driven Auxiliary Feedwater Pump (MDAFP) Flow Control Valves	Other MDAFP **AND** TDAFP Flow Control Valves	Signs on breakers for Flow Control Valves. Signs with flashers on the Flow Control Valves. Tags in Control Room at applicable 101 switches.	Protection of operable Tech Spec equipment. Redundant equipment
Non-Essential Service Water (NESW) Pump	Available NESW Pump	Signs with flasher on pump. Sign on breaker and tag in Control Room at 101 switch.*	Redundant equipment
Plant Air Compressor (PAC)	Opposite Unit PAC **AND** Both CACs **AND** Backup Air Compressor	Signs with flasher on PAC, both CACs and Backup Air Compressor. Sign on breaker and tag in Control Room at 101 switch (except for Backup Air Compressor).*	Redundant equipment
Plant Air Header Ring Isolation Valve, PRV-10, 11, 20, or 21	Remaining PRVs **AND** both Units PACs	Sign on PRV and tag in Control Room at 101 switch. Signs with flasher on PAC.	Protection of plant air ring header.
Reserve Feed	EP Yard (69 kV) **AND** Both Units EDGs	**Note:** Notify FWTDC the Switchyards are guarded. Sign with flasher at EP yard Gate and sign with flasher on EDG doors. Tag in Control Room at 101 switch.	Protection of redundant power supplies