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Operation Reactor Branch No 5
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US Nuclear Regulatory Commission
Washington, DC 20555

DOCKET 50-255 - LICENSE DPR-20 -
PALISADES PLANT - SYSTEMATIC EVALUATION PROGRAM

In recent weeks, extensive discussions have been held between Consumers Power Company and the NRC Staff concerning the Integrated Assessment for Palisades. These discussions have included meetings on January 20 - 22, 1982 and February 4 - 5, 1982, as well as numerous informal conversations over the past several weeks. During those meetings, Consumers Power Company was requested to formally submit our proposals for appropriate action to be taken to address the remaining unresolved findings from the Palisades SEP. The purpose of this letter is to formally submit those proposals.

Attached is a compilation of all SEP Topic difference summaries as transmitted by NRC letter dated January 20, 1982. (Some have been slightly modified based on subsequent discussions with the staff.) Following each difference summary is a brief discussion of the action proposed by Consumers Power Company to resolve the issues, where further action is appropriate. We believe that the actions proposed herein address the underlying NRC concerns in each topic difference summary, and further underscore Consumers Power Company's commitment to work closely with the staff to bring all SEP issues to their final resolution. We believe that these proposals are responsive to the staff, and are responsible solutions for the identified concerns.

You will note that a schedule for each proposed action has not been included in this letter. An expected schedule for each item will be developed and submitted to the staff by separate letter in the near future.

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ATTACHMENT - 16 Pages

SYSTEMATIC EVALUATION PROGRAM
Palisades Plant

Proposed CP Co Action To Resolve Topic Findings

1. II-1.A Exclusion Area Authority and Control

Difference Summary

Potential defects exist in the titles to some lands within the exclusion area boundary which relate to both surface ownership and mineral rights. 10 CFR Part 100.3(a) as implemented by SRP 2.1.7 requires the licensee to have the authority to determine all activities, including exclusion or removal of personnel and property from the area.

Proposed Action

None. CP Co maintains undisputed control over the property. Further legal action is unwarranted.

2. II-3.B Flooding Potential and Protection Requirements

3. II-3.B.1 Capability of Operating Plants to Cope With Design Basis Flooding Conditions

4. II-3.C Safety-Related Water Supply [Ultimate Heat Sink (UHS)]

Difference Summary

The unresolved deviation in the above integrated topics is that the calculated flood level due to a (wind driven wave) seiche, as required by 10 CFR 50 (GDC 2) as implemented by Standard Review Plan 2.4.5, and Regulatory Guide 1.59 is 597.1 feet msl. Flooding of safety-related equipment would occur above 594.67 feet msl.

Proposed Action

None. Recent more sophisticated analyses using current methodology show that a safety margin of approximately six feet exists between a seiche flood level and the limiting safety-related equipment. The 597.1 foot msl level is excessively conservative.

A draft summary report of the current analysis was discussed with the NRC and its consultant at a meeting on February 3, 1982. The final report will be submitted to the NRC approximately March 15, 1982.

5. III-1 Classification of Structures, Components And Systems (Seismic and Quality)

Difference Summary

10 CFR 50 (GDC 1) as implemented by Regulatory Guide 1.26 requires that structures systems and components important to safety be designed, fabricated, erected and tested to quality standards commensurate with the importance of the safety functions to be performed. Category C joints of vessels, which would currently be classified by ASME Section III, 1977 as Class 2 or 3 but built to ASME Section III, 1965 as Class C do not satisfy current radiography requirements.

Proposed Action

Complete the review and verify, on a sampling basis, the adequacy of the open items listed in NRC letter dated December 28, 1981.

6. III-2 Wind and Tornado Loadings

Difference Summary

The requirements of 10 CFR 50 (GDC 2) as implemented by Regulatory Guide 1.117 and SRP 3.3 prescribe structures, systems, and components that should be designed to withstand the effects of a tornado without loss of capability to perform their safety function.

The following structures and components have not been shown to be able to withstand tornado wind loadings:

1. Safety injection and refueling water tank

2. Supply and exhaust piping for emergency diesel generators

3. Steel frame enclosure over the spent fuel pool

4. Condensate Storage Tank

Proposed Action

1. Alternate water sources exist for PCS makeup during cooldown if the SIRW tank is lost. CP Co will review those protected water sources available for PCS makeup and verify that procedures and equipment are sufficient for operators to use those sources in a timely manner.

2. None. The piping from the redundant diesel generators is separated and adequately protected so that common mode failures are extremely unlikely.

3. None. The siding system is expected to be largely stripped off by the wind, making extreme loading and gross failure of the structure

unlikely. The resulting siding missiles would be less severe than others already investigated.

4. Alternate water sources for the Auxiliary Feedwater System are available. CP Co will review those protected water sources available for steam generator makeup and verify that procedures are sufficient for operators to use those sources in a timely manner.

7. III-4.A Tornado Missiles

Difference Summary

10 CFR 50 (GDC 2) as implemented by Regulatory Guide 1.117 prescribes structures, systems, and components that should be designed to withstand the effects of a tornado, including tornado missiles, without loss of capability to perform their safety function.

The following safety-related structures, systems, and components were found to not be protected from tornado missiles:

1. Atmospheric relief stacks of steam relief valves

2. Safety Injection and Refueling Water Tank

3. Compressed air system

4. Diesel generator air intake and exhaust piping

5. Condensate Storage Tank

Proposed Action

1. None. The likelihood of crimping all twelve stacks in their protected location such that none will pass the small volume of steam necessary is extremely remote.

2. As in III-2 above, review protected water sources which are available for PCS makeup and verify that procedures and equipment are sufficient for the operators to use those sources in a timely manner.

3. None. The compressed air (service and instrument air) system is desirable but not required to safely shut down the plant.

4. None. As in III-2 above, the possibility of common mode diesel failures is extremely remote, due to the separation and protection which already exists.

5. As in III-2 above, review protected water sources for steam generator makeup and verify that procedures are adequate for the operators to use those sources in a timely manner.

8. III-5.A Effects of Pipe Break on Structures, Systems and Components Inside Containment

Difference Summary

10 CFR 50 (GDC 4), as implemented by Regulatory Guide 1.46 and SRP 3.6.2, requires in part that structures, systems and components important to safety be appropriately protected against dynamic effects, such as pipe whip and discharging fluids that may result from equipment failures. The effect of pipe breaks inside containment was not a part of the original design basis of the Palisades plant.

Proposed Action

Continue evaluation considering the latest guidance provided in NRC letter dated December 4, 1981. A schedule for any necessary modifications will be determined at the completion of the evaluation.

9. III-5.B Pipe Break Outside Containment

Difference Summary

10 CFR 50 (GDC 4), as implemented by SRP 3.6.1 and 3.6.2, requires in part that structures, systems and components important to safety be appropriately protected against dynamic effects, such as pipe whip and discharging fluids that may result from equipment failures.

Reactor coolant letdown piping break evaluation, the jet expansion model and evaluation of the effects of cracks in seismic Category I moderate-energy lines have not been adequately evaluated by the licensee.

Proposed Action

None. It is believed that additional information submitted to NRC after issuance of the draft SER adequately resolves this topic.

10. III-7.A Inservice Inspection, Including Prestressed Concrete Containments With Either Grouted or Ungrouted Tendons

Difference Summary

10 CFR 50 (GDC 53) as implemented by Regulatory Guide 1.35 in part requires the following:

1. Tendon force acceptance criteria which varies with time

2. Tendon forces be reported rather than wire forces

3. Concrete surrounding the tendon end anchorages be visually inspected during the integrated leak rate tests

The inservice inspection program at Palisades does not meet the items stated above.

Proposed Action

1. Any action which may be appropriate for this issue is still being evaluated.

2. Complete. No further action required.

3. None. Concrete inspections already performed during tendon surveillance tests adequately monitor anchorage condition.

11. III-7.B Design Codes, Design Criteria, and Loading Combinations

Difference Summary

10 CFR 50 (GDC 1, 2 and 4) as implemented by SRP 3.8 requires that structures systems and components be designed for the loading that will be imposed on them and that they conform to applicable codes and standards.

Code changes affecting specific types of structural elements have been identified where safety margins in structures which would be required by current versions of the applicable codes and standards are significantly increased. 22 specific areas of design code changes which may be applicable to the Palisades plant design have been identified where the current code requires substantially greater safety margins than the earlier version of the code or where no original code provision existed.

Proposed Action

CP Co will review the 22 specific changes identified in the NRC evaluation for applicability and determine a method to evaluate the effect of the applicable changes as they pertain to the adequacy of plant design.

12. III-7.C Delamination of Prestressed Concrete Containment Structures

Difference Summary

Standard Review Plan 3.8.1 and the ASME Code, as part of the implementation of GDC 16, state that consideration should be given to radial forces for portions of prestressed containments with curvature. These forces were not considered in the initial design. There is no radial reinforcement in the dome concrete resulting in tension forces being resisted by the concrete and possible delamination. A post-construction method to consider these radial forces is by inspection. There is no inspection program currently being performed by the licensee to assure that delamination has not occurred.

Proposed Action

A post-construction delamination inspection was performed in 1970 with no delamination noted. CP Co will perform an additional one-time inspection of the dome using methods similar to the 1970 inspection and, in addition, agree to include in Technical Specifications a requirement to perform an additional similar inspection in the event that a corrective retensioning program is required for 5% or more of the total number of dome tendons installed.

13. III-8.A Loose Parts Monitoring and Core Barrel Vibration Program

Difference Summary

The requirements of 10 CFR 50 (GDC 13) as implemented by Regulatory Guide 1.133, Revision 1, and SRP Section 4.4 prescribe a loose parts monitoring program for the primary system of light-water-cooled reactors. Palisades does not have a loose parts monitoring program that meets the criteria of Regulatory Guide 1.133.

Proposed Action

None. An LPM system would not significantly improve overall plant safety.

14. V-5 Reactor Coolant Pressure Boundary Leakage Detection

Review Criteria

10 CFR 50 (GDC's 2 and 30) as implemented by SRP 5.2.5 and R. G. 1.45 requires the measurement of leakage from the reactor coolant pressure boundary (RCPB) to the containment and interfacing systems and states design criteria for the systems employed for such.

For systems employed for measurement of leakage from the RCPB to the containment, R. G. 1.45 states that: 1) system should be an airborne particulate radioactivity monitor that is SSE qualified, 2) a minimum of two others should be present which are OBE qualified, and 3) all systems should have a sensitivity to detect leakage of 1 gpm within 1 hour. Those employed for measurement of intersystem leakage should include sensors for things such as radioactivity, flow, level, pressure, temperature, etc. and be OBE qualified. All the above systems should 1) have alarms and indicators in the main control room, 2) be readily testable and calibrated during normal operation, and have their availability in the technical specifications.

Difference Summary

1) The leakage detection systems incorporated for measurement of leakage from the reactor coolant pressure boundary to the containment are not in conformance with Regulatory Guide 1.45 criteria with regard to not being sufficient in 1) number of types of systems, 2) sensitivity, 3) seismic resistance and 4) testability during normal operation.

2) A section is lacking in the Palisades Technical Specifications concerning operability of the reactor coolant pressure boundary containment leakage detection system.

3) Information concerning the leakage detection systems for the detection of inter-system reactor coolant pressure boundary leakage and the CVCS Makeup Flowrate is incomplete. Therefore, we cannot determine the extent to which Regulatory Guide 1.45 is met.

4) The reactor coolant inventory balance is only capable of 1 gpm sensitivity, performed on a daily basis, not 1 gpm w/in 1 hr. as would be required for reliance on this system. In addition, the seismic qualification and testability during normal operation requirements for detection systems are not met. Therefore, it is not appropriate to rely upon this for leakage detection.

Proposed Action

1. Possible action for this topic will be deferred until final action is identified for Topic III-5.A (Effects of Pipe Breaks on Systems, Structures and Components Inside Containment) since local leak detection systems may be needed.

2. Possible action for this topic will be deferred until final action is identified for Topic III-5.A (Effects of Pipe Breaks on Systems, Structures and Components Inside Containment) since local leak detection systems may be needed.

3. Possible action for this topic will be deferred until final action is identified for Topic III-5.A (Effects of Pipe Breaks on Systems, Structures and Components Inside Containment) since local leak detection systems may be needed.

4. None. The PCS inventory balance is already required by Palisades Technical Specifications to be performed at three times the frequency in the CE Standard Technical Specifications. The seismic qualification status of the instruments used has no significant impact on plant safety.

15. V-10.B RHR Reliability

16. V-11.A Requirements for Isolation of High Pressure/Low Pressure Systems

Difference Summary

1) Overpressure relief capacity is required by 10 CFR 50 (GDC's 19 and 43) as implemented by SRP 5.4.7, BTP ASB 5-1, and Regulatory Guide 1.139 for the Shutdown Cooling System (SCS) when in operation, i.e., not isolated from the reactor coolant system. The Overpressurization Protection System (OPS) fulfills this function and is required by Technical Specifications. There is no procedural requirement in the Technical Specifications, however, that assures the OPS is placed in service before the SCS is placed in service. For instance, present procedures place the SCS in service at 325°F and 220 psi, whereas the OPS is placed in service at 300°F and 400 psi. The OPS must be placed in service at 250°F by Technical Specifications.

2) 10 CFR 50 (GDC 19 and 34) as implemented by SRP 5.4.7, BTP ASB 5-1 and Regulatory Guide 1.139 require that the plant can be taken from normal operating conditions to cold shutdown using only safety-grade systems, assuming a single failure and utilizing either onsite or offsite power through the use of suitable procedures. The Palisades plant has safety-grade plant systems capable of safe shutdown under these conditions; however, the plant operating procedures rely upon other non-safety grade systems and have not been verified to specify how the cooldown would be accomplished by the operator in the event of failures in non-safety grade systems.

Proposed Action

1. Change Technical Specifications to require the OPS to be in service whenever the SCS is in service.

2. Review existing procedures to verify that operators are provided with sufficient guidance to direct use of safety grade systems in the event of failures in nonsafety grade systems and make suitable changes if appropriate.

17. V-II.A (Electrical) Requirements for Isolation of High and Low Pressure Systems

Difference Summary

10 CFR 50 (GDC 35) requires that the ECCS have suitable interconnections and isolation. The High Pressure (HP) and Low Pressure (LP) Safety Injection (SI) Systems share common headers with the headers being separated from each other by a single check valve and a motor operated valve in series. The HPSI and LPSI pumps and the motor operated valves

between the headers all start at the same time. In the event of a single check valve failure, some HPSI flow will be diverted to the LPSI system and the LPSI could be overpressurized and damaged. The consequences of this event have not been analyzed by the licensee.

Proposed Action

Leak testing of the check valves of concern is already required by Palisades Technical Specifications. This testing is adequate to verify check valve condition on a routine basis. However, procedural changes will also be made to verify check valve closure prior to criticality after each use of the LPSI system for shutdown cooling.

18. VI-2.D Mass and Energy Release for Postulated Pipe Breaks Inside Containment

19. VI-3 Containment Pressure and Heat Removal Capability

Difference Summary

10 CFR 50 (GDC's 16, 38, and 50) as implemented by SRP 6.2.1.4 require that containment design conditions are not exceeded during an accident assuming a single failure. Failure of a single MSIV results in a two steam generator blowdown, which exceeds containment design pressure by 1.53 times.

(Operational Note: The MSIV check valves at the Palisades plant have failed to close on three occasions following a shutdown and cooldown. On September 21, 1972, CV-0510 failed to operate because the linkages were sticky on six of the solenoid valves. The stickiness was attributed to dirt, and the solenoid valve linkages were cleaned and relubricated. The solenoid valve linkages have been covered with plastic covers to minimize dirt pickup and a dry lubricant, recommended by the manufacturer to limit long term dirt pickup, is applied at every outage.

On May 19, 1973, CV-0501, and on August 12, 1973, CV-0510 failed to close because of the binding in the stuffing boxes. The inside and outside diameter of the lantern rings and packing followers were machined, in accordance with factory recommendations, to increase the clearance around the operating shaft. The valve packing is now inspected at each refueling outage and the MSIVs are exercised several times following each cold shutdown).

All the above failures were under essentially no-flow conditions. Flow tends to close the MSIVs.

Proposed Action

Perform appropriate modifications to make MSIV/main steam system configuration single failure proof with respect to concerns for two steam generator blowdowns inside containment.

20. VI-4 Containment Isolation Systems

Difference Summary

1. The isolation valving arrangements do not meet the requirements of 10 CFR 50 (GDC 55 or 56) as implemented by SRP 6.2.4 from the standpoint of valve location for penetrations 1, 4, 4a, 10, 11, 25, 26, 30, 31, 33, 36, 37, 38, 39, 40, 40a, 41, 42, 44, 45, 46, 47, 49, 52, 65, 67, 68, and 69.

2. Isolation valves differ from the explicit requirements of 10 CFR 50 (GDC 55, 56, and 58) as implemented by SRP 6.2.4 from the standpoint of valve type by using one check valve in series with other type isolation valves located outside containment for penetrations 7, 8, 10, 11, 14, 26, 30, 31, 37, 39, 41, 42, 45, 65, and 67.

3. Isolation barriers differ from the explicit requirements of 10 CFR 50 (GDC 55, 56 and 57) as implemented by SRP 6.2.4 from the standpoint that pipe caps or blind flanges are used as containment isolation barriers as follows:

   a. Penetrations with pipes or test connections capped outside containment: 13, 17, 21, 25, 28, 29, 38, 39, 48, and 73;

   b. Penetrations with blind flanges inside containment: 18, 27, 29, and 73; and

   c. Several lines associated with the following penetrations which are equipped with pipe caps: the personnel air lock (penetration 19); emergency access air lock (penetration 50); and equipment hatch (penetration 52).

4. 10 CFR 50 (GDC 55) as implemented by SRP 6.2.4 requires that two automatic valves be provided for isolation. Penetration 44 shows a manual isolation valve (3/4"-2084) in series with an air operated isolation valve that differs from the requirement from the standpoint of valve actuation.

Proposed Action

1. None. GDC configurations offer no significant improvement in safety over existing plant configuration.

2. None. GDC configurations offer no significant improvement in safety over existing plant configuration.

3. By ASME B&PV Code, 1980 edition, Section III, Subsection NE, Article 3367, threaded caps are allowed for closure of penetrations of 2" diameter and smaller. CP Co will provide additional discussions to address all open items in this paragraph under separate cover.

4. Provide second remotely operated valve in series with the existing isolation valve for Penetration 44.

21. VI-6 Containment Leak Testing

Difference Summary

10 CFR 50, Appendix J requires that tests be performed to assure that leakage through the primary reactor containment and systems and components penetrating primary containment shall not exceed allowable leakage rate values as specified in the technical specifications or associated bases.

The licensee has requested an exemption to the requirements of airlock leak testing if the airlock is opened between the six-month Type B test. This request for exemption has been denied and the licensee has been requested to modify the technical specifications to require a full Type B test of the personnel airlock at a pressure of P_a at least once every six months, with a verification of airlock door seal integrity within 72 hours of each opening or the first of a series of openings during the interim between six month tests, whenever containment integrity is required. Plant modification will be required in order to perform this testing.

Proposed Action

This issue is being addressed outside the SEP as a current licensing issue.

22. VI-10.A Testing of Reactor Trip System and Engineered Safety Features Including Response Time Testing

Difference Summary

10 CFR 50.55a (h) through IEEE Std. 279-1971, Sections 3 (9) and 4.10 requires that response time testing be performed on a periodic basis for construction permits issued after January 1, 1971. At Palisades, the control rods, the diesel generators start time and load sequencers, and the containment isolation valves are response time tested, but overall response time testing is not being performed.

Proposed Action

None. Additional time-response testing beyond that already performed would not significantly improve plant safety.

23. VII-1.A Isolation of Reactor Protection System from Non-Safety Systems, Including Qualifications of Isolation Devices

Difference Summary

10 CFR 50.55a (h) through IEEE Std. 279-1971 requires that safety signals be isolated from non-safety signals and that no credible failure at the output of an isolation device shall prevent the associated protection system channel from meeting the minimum performance requirements specified in the design bases.

At Palisades, electrical signals from RPS steam generator pressure channel B and reactor coolant flow channel A are run to the plant computer with only resistor isolation.

Proposed Action

Suitable isolation devices or channel separation which meet IEEE 279-1971 will be provided for the three circuits of concern.

24. VII-3 Systems Required for Safe Shutdown

Difference Summary

Electrical Items

1. 10 CFR 50 (GDC 17) requires that two paths must be available from the safety busses to the offsite power system. One path must be immediately available and the other path must be made available in a short period of time. At Palisades, it will take four to six hours to establish the delayed access path. (Only two hours of battery capacity exist) and the consequence of loss of all ac and dc have not been evaluated.

2. 10 CFR 50.55a (h) requires that channels that provide signals for the same protective function shall be independent and physically separated to accomplish decoupling of the effects of unsafe environmental factors, electric transients, and physical accident consequences documented in the design basis, and to reduce the likelihood of interactions between channels during maintenance operations or in the event of channel malfunction.

   At Palisades, the instrumentation for high pressure scram signals does not satisfy this requirement because the channels are divided into two pairs, each pair shares a power source and cable routing, the trip is not fail safe on loss of power and the logic is two out of four.

   Because the Technical Specifications permit operation with only two operable channels (if one of the inoperable channels is tripped), there are several scenarios in which the high pressure trips fail.

The most simple of these is to fail the dc source which is assumed to be common to the remaining channels. Alternatively, assuming one of the four reactor protection system channels is bypassed, but not tripped (operating in a two out of three logic arrangement for reactor trip) and assuming the single failure of one raceway or one dc power source causes failure of two high pressure trip signals, the reactor would not trip when required.

3. 10 CFR 50 (GDC 17) requires that both onsite and offsite power systems shall provide power to systems and components important to safety. At Palisades the Boric Acid Injection System is not independent of a loss of offsite power because the Boric Acid Tank heaters (which ensure boron remains in solution) are not powered from a safety related bus (i.e., are not automatically powered from the diesel generator).

4. The component cooling water surge tank level is measured by a single transmitter (LT 0917) with indication provided in the control room. This does not satisfy Section 4.20 of IEEE Standard 279-1971.

5. The output pressure of the component cooling circulating pumps is measured by a single pressure sensor and transmitter (PT-0918) with indications provided in the control room. This does not satisfy Section 4.20 of IEEE Standard 279-1971.

6. 10 CFR 50 (GDC's 2 and 34) as implemented through SRP 5.4.7, BTP RSB 5-1, and R. G. 1.139 in part require that the Seismic Category I water supply for auxiliary feedwater have sufficient inventory to permit operation at hot shutdown for four hours followed by cooldown to conditions permitting shutdown cooling system (SCS) initiation. The inventory is based on the cooldown time assuming a single active failure and either only onsite or only offsite power.

   Sufficient safety-grade water is not maintained in a seismically qualified tank(s) to perform this function.

Proposed Action

(NOTE: The above summaries do not totally reflect the latest CP Co-NRC correspondence.)

1. As discussed in previous correspondence, substantially more battery capacity has now been installed to alleviate this concern. Procedures will be reviewed and modified as necessary, however, to ensure that operators have guidance for removing nonessential dc loads to further extend battery life if conditions warrant.

2. None. This item is no longer an issue for this topic.

3. None. Sufficient redundancy in heaters, power supplies and instrumentation exists so that the existing system is essentially equivalent to current criteria.

4. Provide a second channel of CCW expansion tank level indication.

5. None. Other diverse indications would exist in the event of loss of CCW.

6. Steam generator makeup water is available through a seismically qualified path from the ultimate heat sink (UHS) independent of external plant tanks. Existing procedures will be reviewed to verify that sufficient guidance exists to direct the operator to the UHS source in a timely manner if the normal sources (tanks) were lost.

25. VIII-3.A Station Battery Capacity Test Requirements

Difference Summary

10 CFR 50 (GDC 18) as implemented by Regulatory Guide 1.129 requires periodic testing for determining battery capacity and for demonstrating that the batteries will provide sufficient power under accident conditions. The Palisades program for testing the batteries does not satisfy these requirements.

Proposed Action

Battery capacity and service testing will be implemented for the recently replaced station batteries.

26. IX-3 Station Service and Cooling Water Systems

Difference Summary

10 CFR 50 (GDC 44) requires that for onsite electric power system operation (assuming offsite power is not available) the ultimate heat sink cooling water system safety function can be accomplished, assuming a single failure. With loss of offsite power and the single failure of diesel 1-2, sufficient service water flow may not be provided to prevent exceeding design temperatures in the component cooling water system. The capability exists to throttle service water flow to non-essential components. Procedures do not exist nor have the effects of temperatures in excess of design been evaluated.

Proposed Action

Previous analyses have been very simplistic and are believed to be conservative. A more detailed analysis will be performed to verify that CCW temperature limits are not exceeded for this postulated accident condition. If the analysis indicates a need, procedure changes will be implemented to direct isolation of unnecessary service water loads in this situation.

27. IX-5 Ventilation Systems

Difference Summary

1. The isolation of the engineered safeguard equipment ventilated area remains questionable due to the presence of non-safety grade isolation dampers.

2. The "penetration and fan room" ventilation system performance is vulnerable to failure of either emergency diesel generator. The failure of one diesel to start when required results in loss of either the supply or exhaust fan. The situation could possibly lead to service conditions exceeding the design parameters of equipment housed in these areas.

3. The ventilation equipment for the "Auxiliary and Radwaste Areas," "Turbine Building," "Intake Structure," and "Viewing Gallery, Switchgear and Cable Spreading Areas," service equipment deemed essential for safety. However, these ventilation systems are neither safety grade, powered from emergency sources nor single failure proof.

Proposed Action

1. CP Co will review the revised topic SER when issued and determine at that time whether additional action may be appropriate.

2. CP Co will review the revised topic SER when issued and determine at that time whether additional action may be appropriate.

3. CP Co will review the revised topic SER when issued and determine at that time whether additional action may be appropriate.

28. XV-2 Spectrum of Steam System Piping Failures Inside and Outside Containment

Difference Summary

For analysis of a spectrum of steam line breaks, 10 CFR 50 (GDC's 17, 21 and 35) as implemented by SRP 15.1.5 require that the most severe single active component failure should be assumed and the effect of loss of offsite power should be considered.

A single failure of the main steam isolation valve (MSIV) could cause both steam generators to blow down. This event has not been analyzed for its core performance effects. Other single failures in mitigating systems have not been analyzed to a sufficient extent so that it can be concluded that the effects of the worst single failure have been considered. These other single failures are:

1. Diesel generator failure (with loss of offsite power)

2. Failure of main feedwater isolation

For some of these events, the licensee has recently submitted analyses.

Proposed Action

1. The MSIV/MSS configuration will be modified to make the MSIV/main steam system single failure proof with respect to concerns for two steam generator blowdowns.

2. Analyses which address diesel generator failures have been submitted.

3. It is our understanding that concerns about failures of feedwater to isolate in the event of a steam line break are a current licensing issue being addressed outside the SEP.

29. XV-12 Spectrum of Rod Ejection Accidents

Difference Summary

10 CFR Part 50 (GDC 28) as implemented by Regulatory Guide 1.77 and SRP 15.4.8 require that reactivity limits be established on the reactivity control system. Our analysis, required to demonstrate the acceptability of the reactivity limits, mainly the rod ejection accident, was evaluated for fuel melting (i.e., less than 200 cal/gm) but not fuel cladding failures. The analysis of the number of pins that would experience DNB for the limiting rod ejection event, and the effect on the dose calculations have not been performed.

Proposed Action

CP Co does not believe that a reanalysis would provide significantly different results and would not warrant the expense. CP Co will address this subject further in a separate letter.