U.S. NUCLEAR REGULATORY COMMISSION

REGION III

Report No. 50-255/90020(DRS)

Docket No. 50-255 License No. DPR-20

Licensee: Consumers Power Company
1945 West Parnell Road
Jackson, Michigan 49201

Facility Name: Palisades Nuclear Generating Plant

Inspection At: Palisades Site, Covert, Michigan

Inspection Conducted: September 4, 1990 through March 28, 1991

Inspectors: C. L. Vanderniet for 3/29/91
V. P. Lougheed Date

C. L. Vanderniet for 3/29/91
J. D. Monninger Date

Approved By: C. L. Vanderniet 3/29/91
C. L. Vanderniet, Chief Date
Operations Program Section

Inspection Summary

Inspection on September 4, 1990, through March 28, 1991 (Report No. 50-255/90020(DRS))

Area Inspected: Special safety inspection by regional inspectors and staff members of the Office of Nuclear Reactor Regulation to review allegations concerning (1) potential post loss of coolant accident (LOCA) return to criticality due to insufficient boron in the safety injection tanks, and (2) potential post-LOCA hydrogen burn due to increased amounts of aluminum inside the primary containment.

Results:
No violations were identified. Both allegations were partially substantiated. Regarding the first issue, the NRC determined that a return to criticality could not be ruled out based on realistic assumptions; however, this return would not significantly impact the course of post LOCA recovery from a safety standpoint. Regarding the second issue, the NRC determined that the maximum hydrogen concentration values present in the FSAR were incorrectly low; however, the correct value would still be below flammability limits. Two Unresolved Items were identified and are discussed in sections 2.a and 2.b of the report.

DETAILS

1. Persons Contacted

Consumers Power Company

*K. M. Haas, Reactor Safety Development Manager
*J. L. Kuemm, Plant Licensing Administrator
*T. A. Buczwinski, Reactor Engineering Superintendent
*W. L. Roberts, Licensing Engineer
*G. C. Pratt, Senior Reactor Engineer
*G. F. Packard, General Reactor Engineer
R. J. Gerling, Accident and Transient Analysis Supervisor
D. J. VandeWalle, Director, Safety and Licensing

U. S. Nuclear Regulatory Commission

*M. P. Phillips, Chief, Operations Branch
*C. L. Vanderniet, Chief, Operational Programs Section
*J. K. Heller, Senior Resident Inspector
+B. E. Holian, Project Manager, Office of Nuclear Reactor Regulation

*Denotes those attending the exit meeting on March 28, 1991.

+Denotes those participating in the exit meeting by telephone on March 28, 1991.

2. Allegation (RIII-90-A-0062)

On June 15, 1990, Region III received two allegations regarding the performance of the Palisades facility following a large break loss of coolant accident (LOCA). The first allegation involved the potential for a return to criticality during a LOCA due to insufficient boron in the Safety Injection Tanks (SITs). The second concern pertained to the amount of hydrogen generated during a LOCA, with an emphasis on the potential for a hydrogen burn.

On September 4, 1990, the inspectors met with the alleger and the alleger's attorney. The purpose of this meeting was to obtain a detailed description of the allegations and to receive copies of the alleger's calculations supporting the allegations.

On September 6, 1990, the inspectors visited the Palisades site to collect information regarding the allegations, licensee calculations, and other supporting documentation.

On September 20, 1990, all of the accumulated information was submitted to the Office of Nuclear Reactor Regulation

(NRR) for review and evaluation. NRR was requested to provide a technical evaluation of the allegations. During the course of NRR's review, both the alleger and the licensee were contacted to provide additional information when required. The final NRR safety evaluations are provided as Enclosure 2, "Safety Evaluation on the Potential for a Return to Criticality Following a Large Break LOCA at Palisades," and Enclosure 3, "Safety Evaluation Regarding the Post-LOCA Hydrogen Analysis."

The following paragraphs separate the major allegations into their constituent parts in order to address each specific concern expressed by the alleger. At the end of each section, an overall conclusion of the major allegation is provided.

a. Post LOCA Return to Criticality (partially substantiated)

Synopsis of Allegation: For a number of years Palisades Nuclear Plant has had problems with in-leakage of primary coolant system (PCS) water into the SITs due to leakage past the SIT check valves. In the event of a LOCA, this relatively unborated water would reach the core first, and, combined with PCS water remaining in the core following the accident, result in a return to criticality. No credit should be given for the control rods in this scenario, as they were never shown to insert following a LOCA. In addition, the licensee analyses incorporates improper assumptions which are designed to maximize the conditions for fuel temperature, but are improper to bound the condition of return to criticality.

NRC Review: The individual portions of the issues discussed above are addressed as follows:

(1) Allegation: The SITs are the only method of shutting down the reactor following a large LOCA, as the control rods were never proven to insert during the LOCA.

NRC Review: This portion of the allegation is a restatement of the licensee's assumptions contained in the Final Safety Analysis Report (FSAR). As discussed in Enclosure 2, no credit was assumed, under worst case conditions, for control rod insertion during a large LOCA. Reactor shutdown was to be obtained by the infusion of boron from the SITs and the safety injection systems, which obtain water initially from the borated safety injection refueling water (SIRW) tank.

It should be noted, however, that Palisades is not unique in its treatment of control rod insertion during a large LOCA. The lack of credit for the control rods is a generic assumption at pressurized water reactors (PWRs) built by either Westinghouse or Combustion Engineering.

(2) Allegation: At Palisades, borated water is delayed from reaching the core due to valve in-leakage which allows PCS water to enter the tanks and fill the SIT injection lines.

NRC Review: This portion of the allegation was evaluated from the standpoint that if valve in-leakage were occurring, borated water would be delayed from reaching the core, and as such was substantiated. As stated in Enclosure 4, the SITs do not have recirculation capability and a large volume of water is required to be drawn from the tanks in order to achieve a representative sample. Based on this, the inspectors determined that some stratification was occurring in the SITs. Therefore, the lower boron concentration PCS water would be injected first during the large break LOCA, with a gradual increase up to the full boron concentration SIT water.

(3) Allegation: Palisades has experienced problems with the SIT valves since at least 1982, and it has not been corrected. This results in PCS in-leakage into the SIT. No solution has been found nor any evaluation done of the impact on safety of the plant in the event of an accident.

NRC Review: This portion of the allegation was partially substantiated. Palisades has experienced problems with the SIT valves and PCS in-leakage. However, the licensee has developed an enhanced tracking, sampling, and maintenance program to ensure that technical specification (TS) requirements would be met. In addition, the NRC performed an evaluation regarding the impact on safety of the plant in the event of an accident when SIT in-leakage were occurring.

Based on reviews of plant records, including event reports, deviation reports, and work requests from 1982 until the present, as well as discussions with plant personnel, Palisades has had recurrent problems with valve leakage on the SIT lines. However, the licensee has an extensive tracking, sampling, and maintenance program to ensure that SIT leakage is monitored and technical specification requirements maintained. As a result of these programs the licensee has experienced periods of time when no in-leakage occurred. An increased band for SIT level measurements was requested by the licensee and approved by the NRC in TS amendment No. 136.

An evaluation of the impact of valve leakage on plant safety was performed by the NRC after the allegations had been made, and is discussed in Enclosure 2.

(4) <u>Allegation:</u> The licensee is responsible to provide calculations for the worst possible condition accident scenario to ensure that the plant would be safely shutdown during such a postulated accident. This was not done at Palisades and, although the problem existed for a very long time, it was never properly addressed.

<u>NRC Review:</u> This portion of the allegation was not substantiated. 10 CFR 50.46 and 10 CFR Part 50, Appendix K, impose certain requirements for a licensee to use when performing large LOCA accident analyses. The licensee correctly followed these requirements in the performance of its analyses.

These assumptions, while conservative to show that long term core integrity could be maintained, do not address the alleger's concerns of a return to criticality. However, as noted in enclosure 2, a return to criticality does not necessarily impose additional safety concerns on the plant. The potential for Palisades to return to criticality subsequent to a large break LOCA would not result in prompt criticality due to the effect of void distribution expected during reflood and a short-term return to criticality was not likely to significantly impact the course of LOCA recovery because of the negative feedback of effects of voiding, the cooling effect of increased steaming, and the imminent shutdown from continued injection of high boron concentration ECCS water. Therefore, the return to criticality is not considered a significant safety concern or the worst possible accident scenario.

(5) Allegation: The basic LOCA analysis was performed to magnify fuel temperature and its assumptions in the model do not realistically show the actual amount of PCS water left in the reactor vessel following the blowdown phase of a LOCA, because this water is superficially subtracted in the model to maximize the temperature of the fuel. Also other numerous conservative assumptions for maximizing fuel temperature are done, which are not conservative for calculations of boron in the core following a large break LOCA.

NRC Review: This portion of the allegation was substantiated. The Palisades design basis LOCA analysis correctly complied with the requirements of 10 CFR 50.46 and 10 CFR Part 50, Appendix K. These regulatory requirements, which are concerned with peak cladding temperature, imposed conservative assumptions to maximize the likelihood of core damage in order to show that long term core integrity could be maintained. As noted above, the independent NRC evaluation concluded that a return to criticality could not be ruled out, but that such an event would not significantly impact the course of LOCA recovery.

(6) Allegation: The Palisades large break LOCA analysis did not account for dilution of the Safety Injection Lines.

NRC Review: This portion of the allegation was substantiated. As discussed in item (5) above, the licensee's design basis LOCA analysis complied with 10 CFR 50.46 and 10 CFR Part 50, Appendix K, in showing that long term core integrity was maintained. These calculations did not address the possibility of a return to criticality due to insufficient boron being injected. The design basis analyses also did not address the fact that the SIT valve leakage would have caused a large quantity of water at PCS boron concentration to be injected prior to the higher SIT boron concentrations. The failure to address these concerns in the design basis LOCA analysis was not contrary to regulatory requirements for the analysis, and a realistic boron concentration evaluation, as stated above, did not identify a significant safety concern.

(7) Allegation: The beginning of cycle (BOC) condition, which is the most limiting, was not considered in the 1982 basis analysis.

NRC Review: This portion of the allegation was substantiated. The 1982 analysis was specifically performed for the situation occurring at the end of cycle 5, which was an end of cycle (EOC) condition. However, at the time this analysis was prepared and approved, it was not intended to be applied on a generic basis, but only for the conclusion of cycle 5. A similar analysis was submitted to the NRC to support a TS change request in 1990. The NRC never acted on that request or evaluated the licensee's analysis. The request was subsequently withdrawn by the licensee.

(8) Allegation: The critical boron concentration neutronic calculations used interpolations between the transient analysis values. This neglected the effects of the diluted water in the SIT lines due to valve leakage, which directly affects the outcome of the neutronic calculations.

NRC Review: As noted in the review of sub-allegations 5 and 6 above, the NRC concluded that the licensee's return to criticality analysis continued to use design basis assumptions, which the NRC noted were non-conservative from a criticality perspective. Therefore, this portion of the allegation was substantiated. However, the licensee's analyses were performed in accordance with NRC guidance to address the requirements of 10 CFR 50.46 and 10 CFR Part 50, Appendix K.

(9) Allegation: During the years, alternatively different tanks (two at a time) had in-leakage problems. Therefore, it is conceivable that three or four of the SITs could experience in-leakage at the same time.

NRC Review: This concern was partially substantiated. A review of Palisades records, and conversations with plant personnel confirmed that tank in-leakage has previously occurred on more than one tank at a time. Additionally, because of common lines between the tanks, the lines leading to all four tanks could be at PCS concentration. However, based on past history and the licensee's proactive efforts to monitor and correct valve leakages, the number and extent of tank leakage is decreasing. Therefore, it is considered unlikely that in-leakage into all four tanks would occur simultaneously now or in the future.

Overall allegation conclusion: The allegation was partially substantiated. The alleger's concern that the licensee's SIT boron concentration may not prevent a return to criticality could not be ruled out. However, as stated in Enclosure 2, the decision on the validity of the allegation does not affect the continued operation, or overall safety of the Palisades plant because of the following considerations: (1) the initiating event is a low probability event; (2) the return to criticality requires a failure of all of the control rods to insert prior to reflood; (3) a short term return to non-prompt criticality will not significantly impact the course of LOCA recovery because of the negative feedback of effects of voiding and the cooling effect of increased steaming, and the imminent shutdown from continued injection of high boron concentration ECCS water; (4) a prompt critical condition is unlikely due to the effect of the void distribution expected during reflood; and (5) the licensee's analyses were performed consistent with the requirements of 10 CFR 50.46 and 10 CFR Part 50, Appendix K.

The inspectors have identified concerns with the licensee's design control processes and corrective actions processes. The licensee's Quality Assurance (QA) Program, as described in Topical Report CPC-2A, Revision 10, titled "Quality Assurance Program Description for Operational Nuclear Power Plants" specifies the method by which the licensee implemented the requirements of 10 CFR Part 50, Appendix B at the time these issues were identified to the licensee.

Paragraph 3.2.2 of this topical report specifies that "Errors and deficiencies in approved design documents, or in design methods (such as computer codes) that could adversely affect structures, systems, and components are documented. Action is taken to assure that errors and deficiencies are corrected."

Paragraph 16.2.3 of the topical report specifies that "For significant conditions adverse to quality, necessary corrective action is promptly determined and recorded. Corrective action includes determining the cause and extent of the condition, and taking appropriate action to preclude similar problems in the future."

These concerns will be tracked as an Unresolved Item (255/90020-01(DRS)), pending further review by the NRC staff.

b. <u>Post-LOCA Hydrogen Generation (partially substantiated)</u>

<u>Synopsis of Allegation:</u> The licensee severely underestimated the amount of aluminum in containment, and the analyses performed to support Chapter 14.22 of the FSAR was incorrect and contained several errors. This could result in the hydrogen concentration exceeding the 4.1% flammability limit, which would be catastrophic.

<u>NRC Review:</u> The individual portions of the issues discussed above are addressed as follows:

(1) <u>Allegation:</u> The amount of aluminum insulation inside containment was incorrectly low.

<u>NRC Review:</u> This portion of the allegation was substantiated. The alleger calculated an area of 181,613 square feet of aluminum inside containment. The FSAR reports an area of 152,462 square feet. The licensee, in its independent review of the alleger's calculation, corrected the FSAR value with two increases: one of approximately 27,000 square feet, and one of 11,000 square feet. These increases resulted in a total amount of aluminum of approximately 190,000 square feet. This was larger than the value calculated by the alleger. However, during the 1990 steam generator replacement outage, much of the aluminum insulation on the steam system piping was replaced with a different (fiberglass) type. This would lower the overall amount of aluminum inside containment. Because the licensee agreed that the FSAR value was incorrect, and no accurate total was available, the alleger's calculated amount of aluminum was used in the NRC calculations, as documented in Enclosure 3.

The addition of aluminum or zinc to containment during modifications would have been performed utilizing the requirements of 10 CFR 50.59. These requirements specify that an evaluation be conducted to determine, among other things, whether an increase in the probability or consequences of an analyzed accident could occur. In this case, the increase in aluminum insulation should have been evaluated to determine its impact on maximum hydrogen concentration. The NRC did not review modifications that would have resulted in changes to the amount of aluminum or zinc to containment to determine if the associated safety evaluations were acceptable. This determination is considered an Unresolved Item (255/90020-02(DRS)).

(2) Allegation: No proper correction was administered to the corrosion rates at the beginning of the corrosion process, which were assumed to be flat over a long period of time, but in fact the corrosion rates are very high initially and are changing fast subsequentially.

NRC Review: This portion of the allegation was partially substantiated. As stated in Enclosure 3, the FSAR underpredicted the amounts of aluminum corroded in the first 40 hours, but after that time, the FSAR values were more conservative than those based on time dependent rates. However, the NRR analyses used the alleger's corrosion rates to calculate the maximum hydrogen concentrations.

(3) Allegation: The hydrogen from zinc and galvanized surfaces reaction with water is not fully accounted for in the total hydrogen production.

NRC Review: This portion of the allegation was partially substantiated. NRR, as documented in Enclosure 3, noted that the values of the corrosion of zinc from galvanized surfaces was identical to that given in ORNL-TM-2412, "Design Consideration of Reactor Containment Spray Systems. Part III The Corrosion of Materials in Spray Solutions." NRR also found the corrosion rates of zinc paints used by both the licensee and the alleger to be less than that from recent test programs. In performing its independent analyses, NRR used the alleger's values and concluded that, overall, the hydrogen production rate from zinc paints and galvanized surfaces was a relatively insignificant term in the overall total.

(4) Allegation: The partial pressures and temperatures correction of containment atmosphere do not seem to be included [in the FSAR analysis]. Additionally, no correction on the recombiner intake as a function of temperature and pressure of the containment seems to be included.

NRC Review: This portion of the allegation was partially substantiated. NRC noted, in Enclosure 3, that the original FSAR analysis was probably outdated. However, the licensee's independent review of the alleger's concern used the COGAP computer program (NUREG/CR-2847, "COGAP: A Nuclear Power Plant Containment Hydrogen Control System Evaluation Code") which properly accounted for containment partial pressures and temperatures. The COGAP program, which was approved by NRC, was the one also used in the NRC's independent analysis.

(5) Allegation: The analysis should correct for the sprayed volume of containment, which constitutes using the entire volume].

NRC Review: This portion of the allegation was not substantiated. In Enclosure 3, NRR concluded that the entire free volume of the containment should be used because of the turbulent mixing generated by the break flow jets, containment sprays, and natural convection flows.

(6) Allegation: The differential equation governing production and consumption of hydrogen following the recombiners initiation was never solved and the COGAP program was not utilized either.

NRC Review: This portion of the allegation was partially substantiated. As stated in item (4) above, and in Enclosure 3, the original FSAR solution may have been incorrect. However, the licensee's independent review analysis did use the COGAP computer program, which does properly solve the differential equation regarding production and consumption of hydrogen.

(7) Allegation: Most important is the fact that the Palisades FSAR Section 14.22 allows the plant to reach the control limit of 3.6 volume percent (v/o) hydrogen before initiation of recombiner operation. Because the recombiner, by its nature of small hydrogen intake, does not reduce immediately the hydrogen content of the containment atmosphere, its action is not sufficient to bring the hydrogen concentration down immediately, and, in fact, the concentration would be still rising to the level much higher than 4.1% flammability limit because of the continuous production of hydrogen from all the other sources. This, obviously, could be catastrophic.

NRC Review: This portion of the allegation was partially substantiated. NRR, in Enclosure 3, agreed that the FSAR did state that the hydrogen recombiner did not need to be actuated until a limit of 3.6 v/o hydrogen was reached. However, this was contradicted by the plant's emergency operating procedures (EOP) which require one recombiner to be started whenever containment pressure exceeds 3.7 psig. Containment pressure would reach 3.7 psig within a few seconds following a large LOCA. The staff concluded that the FSAR did require updating in order to reflect the EOP guidance, but that the recombiners would be started within a reasonable time frame following a LOCA. Furthermore, the independent NRC calculation of post-LOCA hydrogen concentrations produced a maximum hydrogen value of 2.9% occurring in approximately 8 days. This is sufficiently below the flammability limit.

Overall allegation conclusion: This allegation was partially substantiated. The staff agreed that the licensee's FSAR Section 14.22 was outdated and needed to be revised, and noted that the licensee had been in the process of updating this section of the FSAR prior to the time the allegation was received by the NRC. An independent analysis of the post-LOCA hydrogen concentration calculated using the NRC approved COGAP computer program showed the maximum hydrogen concentration to be 2.9% peaking within 8 days. Therefore, the alleger's conclusion that hydrogen flammability limits would be reached, with the potential for a hydrogen burn, was not substantiated.

3. Review of Additional Information

The NRC staff has factored any and all additional technical information received by March 22, 1991, from either the alleger or licensee, into its final conclusions. Any other concerns expressed by the alleger are undergoing further review, and will be addressed in separate correspondence.

4. Unresolved Items

Unresolved items are matters about which more information is required in order to ascertain whether they are acceptable items, items of non-compliance, or deviations. Unresolved items disclosed during the inspection are discussed in Paragraphs 2.a. and 2.b.(1) of this report.

5. Exit Interview

Discussions were held with both licensee representatives and the alleger by NRC staff throughout the inspection. An exit meeting was held at the site on March 28, 1991. The inspectors summarized the scope and findings in regard to the allegations. The inspectors also discussed the likely informational content of the inspection report. The licensee did not identify any such content as proprietary.

Enclosure 2

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION RELATING TO AN ALLEGATION REGARDING THE POTENTIAL FOR A RETURN TO CRITICALITY FOLLOWING A LARGE BREAK LOCA AT PALISADES

CONSUMERS POWER COMPANY

PALISADES NUCLEAR PLANT

DOCKET NO. 50-255

1.0 INTRODUCTION

By letter of September 20, 1990 (Ref. 1), Region III requested NRR to assist in review of an allegation regarding LOCA analyses for Palisades. In References 1 through 3, the alleger indicated that the assumptions which are intended to assure a conservative estimate of fuel cladding temperature are not conservative with respect to the potential for post-LOCA criticality. We have reviewed the allegation based on the information in References 3 and 4 and additional submittals provided by the alleger during the period of October 1990 to January 1991. We evaluated the information provided by the licensee in conference calls and follow-up submittals to the NRC. In addition, we have examined LOCA test data to assist in review of the allegation. As a result, we have prepared the following evaluation.

2.0 EVALUATION

The Palisades LOCA analyses are performed using analysis assumptions specified in Appendix K to 10 CFR 50. These assumptions are defined with the deliberate intention of assuring a conservative calculation of peak cladding temperature. Some of the more important assumptions in this regard include: (1) use of the Moody model to maximize the blowdown rate, (2) loss of all safety injection water to the containment during the bypass period, and (3) an assumed loss of offsite power with consideration of single failure to minimize the safety injection flow.

The effect of these assumptions is to minimize the post-blowdown water inventory in the reactor pressure vessel and thereby prolong the time period during which the core is uncovered. This results in a conservatively high estimate of peak cladding temperature. It also results in the core being reflooded with water only from the highly borated safety injection tanks (SIT) or safety injection system (SIS). Reactor shutdown is easily achieved under these assumptions (because of the combination of boron and voiding). No credit is assumed for control rod insertion.

From a reactivity perspective however, these same assumptions estimate a less reactive state than would be expected in a more realistic scenario. In a realistic scenario significantly more primary system water would remain in the vessel following the vessel blowdown. This water would be cool (relative to normal operating temperature) and would be at the full power boron concentration. This residual water would dilute (in boron concentration) the incoming safety injection water and thereby significantly alter (in the nonconservative direction) the reactivity potential of the mixture. In addition, water remaining in the vessel will result in an earlier start of reflooding since less injection water is needed to fill the lower plenum. This means that fuel rod temperatures will be lower (than in the LOCA analyses) and will result in less boil off during the initial reflood stages. Since boil off leads to an increase in boron concentration of the remaining coolant and increased voiding, this too is nonconservative from a reactivity perspective. The more low boron concentration water remaining in the reactor vessel the more likely that reflood will result in a critical core. Thus, the amount of the remaining water in the reactor vessel has a significant effect on the reactor remaining shutdown following a large break LOCA.

In reviewing the allegation, we have examined the licensee's post-LOCA criticality analyses documented in References 3 and 5, and additional calculations of October 25, October 13, and November 27, 1990. In these analyses the licensee has attempted to demonstrate that the boron concentration in the vessel water mixture at the time of reflood is greater than the critical boron concentration. Our effort included a review of the Palisades plant primary coolant system (PCS) and ECCS volumes and configuration, model and analysis assumptions, system response. We also examined the amount of reactivity which could be added to the system by the cooldown of the coolant during depressurization. The review found that a cooldown of 300°F provides the potential for a very significant reactivity insertion (several percent). We also determined that the licensee's conclusions regarding recriticality depend upon assumptions and outputs from the licensing basis LOCA calculations. As discussed above, design basis LOCA analyses minimize the amount of low boron concentration water remaining in the vessel and are therefore nonconservative from a criticality perspective.

Finally, we examined experimental data pertinent to the question. LOFT test L1-2 described in Reference 6 is a large break blowdown test conducted from typical PWR initial temperature and pressure. The test result shows that water equivalent to 90 percent of water volume of the lower plenum remains in the reactor vessel after the blowdown. While the applicability of this data to Palisades may be debated, it nonetheless is a data point based upon simulated PWR conditions. For a smaller break LOCA (but large enough to discredit control rod insertion for reactor shutdown) it may be expected that more water could remain in the reactor vessel after the blowdown because of a smaller depressurization rate.

3.0 DISCUSSION

Based on the above considerations it is the staff's position that a return to critical cannot be excluded. However, it is expected that only a prompt critical configuration would be of concern from a safety perspective. This is considered unlikely because of the need for a very rapid reactivity insertion.

Experimental data (Ref. 7) indicates that during reflood when the coolant comes in contact with hot fuel rods, significant voiding occurs. This has the beneficial effect of introducing negative reactivity from the increase in voids, as well as reducing the rate of reflooding (and thus the rate of addition of positive reactivity) due to the increased resistance of the two phase mixture. A non-prompt return to critical would not be expected to have significant effect on the plant recovery following a LOCA because these same negative feedbacks would limit the power level, and increased steaming would assist in cooling those portions of the core which are uncovered. In addition, the return to power would be short-lived because of the continued boron injection from the ECCS. These factors, in conjunction with the low likelihood of the initiating event (large break LOCA), and the consideration that the control rods must also fail to insert lead to the conclusion that continued operation is acceptable.

4.0 CONCLUSION

Based on our review of the alleger's submittals, the licensee's submittals, and the test data discussed in Section 2.0, we conclude that a return to criticality cannot be excluded. This conclusion is based upon the following: (1) there is the potential for a significant reactivity insertion due to cooldown of the moderator during depressurization; (2) test data indicate that large amounts of (low boron concentration) primary system water may remain following the blowdown of the primary system; and (3) the licensee's criticality analyses use design basis LOCA models and assumptions which minimize the amount of low boron concentration water involved in the core reflood and are thus non-conservative relative to a return to criticality. However, our decision on the validity of the allegation in this SER does not affect the continued operation of the Palisades plant. This is based on the following consideration:

(1) The initiating event (large break LOCA) is a low probability event;

(2) A return to criticality requires failure of the control rods to insert prior to reflood;

(3) A short term return to non-prompt criticality is not likely to significantly impact the course of LOCA recovery because of the negative feedback of effects of voiding and the cooling effect of increased steaming (if a notable power generation level is achieved), and the imminent shutdown from continued injection of high boron concentration ECCS water;

(4) A prompt critical condition is unlikely due to the effect of the void distribution expected during reflood.

(5) The licensee's analyses have been performed consistent with the requirements of 10 CFR 50.46 and Appendix K to 10 CFR 50.

5.0 REFERENCES

1. Letter from H. J. Miller, to Dennis M. Crutchfield, "Request for Assistance Allegations Regarding the Palisades Nuclear Plant," dated September 20, 1990.

2. Attachment 1 of Reference 1 - Transcript of September 4, 1990 meeting between the alleger, Region III, and NRR (via telephone).

3. Attachment 2 of Reference 1 - Alleger File on Safety Injection Tank Boron Concentration.

4. Attachment 5 of Reference 1 - Consumers Power Company letter dated August 1, 1988 to the NRC, Concerning a Tech Spec Change Request for Modification of Peaking Factors and LOCA Limits.

5. Attachment 8 of Reference 1 - Safety Injection Tank Boron Analysis from 1982, 1985, 1986, and the 1982 licensing submittal.

6. TREE-NUREG-1026, "Experimental Data Report for LOFT Nonnuclear Test L1-2," January 1977.

7. JAERI-M-90-236, "Evaluation Report on SCTF Core-II Test S2-08 (Effect of Core Inlet Subcooling on Thermal-Hydraulic Behavior Including Two-Dimensional Behavior in Pressure Vessel During Reflood in PWR-LOCA)," January 1991.

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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

PLANT SYSTEMS BRANCH

POST-LOCA HYDROGEN ANALYSIS

PALISADES NUCLEAR PLANT

DOCKET NO. 50-255

1.0 INTRODUCTION

By memorandum of September 20, 1990 (Reference 1) Region III requested NRR's assistance in responding to an allegation regarding the Palisades Nuclear Plant. The Plant Systems Branch of NRR has reviewed the allegation concerning the hydrogen generation following a Loss of Coolant Accident (LOCA). The Materials and Chemical Engineering Branch has provided input (Reference 9) on the corrosion issue.

The alleger, who was employed as a design engineer in the transient analysis group for the facility, indicated that the hydrogen analysis in Section 14.22 of Palisades FSAR has many errors, and specifically, identified the following problem areas:

1. amount of aluminum insulation,
2. corrosion rates,
3. hydrogen generation from zinc and galvanized surfaces,
4. correction of partial pressures and temperature in containment atmosphere,
5. correction of free volume from effective spray volume,
6. correction of recombiner intake as a function of temperature and pressure,
7. initiation time of hydrogen recombiner, and
8. improper solution for the differential equation governing production and consumption of hydrogen following the initiation of recombiners.

Particularly, the alleger has a grave concern on the consequences of Item No. 7; the word "catastrophic" was used by the alleger concerning this issue. The above allegations were mainly based on the results of a calculation performed by the alleger for the facility and documented in Reference 2. The original FSAR analysis was performed in 1976.

Our initial review indicated that some of the allegations might be valid and could have some impact on plant safety. To evaluate the significance of the impact, we requested our consultant at Sandia National Laboratories to perform an independent analysis using the computer code, COGAP. The COGAP code, which is described in Standard Review Plan Section 6.2.5, was developed and has been used by the NRC staff to calculate hydrogen concentration following a LOCA. The results of Sandia's analysis show that a maximum hydrogen concentration of 2.9 volume percent (v/o) will occur 8 days into a LOCA.

2.0 EVALUATION

The staff reviewed each of the problem areas identified by the alleger and our evaluation is provided below. The review was based on Regulatory Guide (RG) 1.7 and Standard Review Plan (SRP) 6.2.5. The basic safety concern is whether the containment hydrogen concentration could exceed 4.0 v/o following a LOCA.

In addition to the review, the staff's consultant at Sandia performed an analysis to evaluate the overall safety significance of the above allegations. To prepare a set of conservative input for the Sandia's COGAP analysis in response to the allegations, the staff carefully evaluated the FSAR data, the alleger's analysis (Reference 2), the licensee's review of alleger's analysis (Attachment 15 to Reference 1), and RG 1.7

2.1 The amount of aluminum insulation

The total aluminum surface area documented in Table 14.22-4 of the FSAR is 152,462 ft^2. The aluminum surface area, calculated by the alleger in Reference 2, is 183,613 ft^2. Apparently, the number used in the FSAR is not conservative.

The staff called the licensee on November 27, 1990, to verify the data. The licensee was aware that the FSAR might need changes and was in the process of re-evaluating the FSAR analysis. Before the completion of the re-evaluation, the licensee performed a review analysis (Attachment 15 to Reference 1) using the computer code COGAP and input data from the alleger. The results of this analysis indicated that the maximum hydrogen concentration was less than 4 v/o, but was much higher than was documented in the FSAR. The licensee used this analysis to demonstrate that there had not been any immediate safety concern. However, the licensee had not decided whether all the assumptions used by the alleger were correct and whether the FSAR needed to be revised. The licensee originally planned to complete its reanalysis by the end of 1990, but the schedule was slipped to the time prior to plant restart.

The COGAP computer code, which is described in SRP 6.2.5, has been used by the staff to calculate hydrogen concentration following a LOCA. Therefore, it is acceptable. In reviewing the licensee's COGAP analysis (Attachment 15 to Reference 1), however, we found that several input parameters used by the licensee were inconsistent with RG 1.7. Therefore, the staff requested Sandia to perform an independent analysis. Since the licensee did not provide any additional data on the amount of aluminum, the conservative number (183,613 ft^2) from the alleger was used as an input to Sandia's analysis.

2.2 Corrosion rate of aluminum

Based on Reference 3, the alleger pointed out that the corrosion rates should be a function of time and that the FSAR assumed the rates to be constant for a long period of time.

The most significant factors affecting corrosion rate of aluminum are time, temperature, and pH value. Review of Reference 3 has indicated that there are experimental data on the growth of aluminum corrosion film with time. Further, aluminum corrosion rate is a very sensitive function of temperature and pH values. The pH value of the chemically treated containment building spray in the Palisades plant is around 7, which is consistent with the value used in the FSAR. The corrosion rates in the FSAR are temperature dependent, but only three temperature points are given.

The staff compared the aluminum corrosion rates between the FSAR and Reference 3. Further, using the temperature values as a function of time, we calculated the amount of aluminum corroded as a function of time. The results shown in Figure 1 indicate that for the first 40 hours after an accident, the FSAR methodology underpredicts the amounts of aluminum corroded, but after that time the FSAR results are more conservative relative to the results based on the time dependent corrosion rates of Reference 3.

In light of the time (8 days) that the maximum hydrogen concentration occurs, the FSAR corrosion rates for aluminum probably generate more hydrogen than the alleger's corrosion rates do. However, to evaluate the impact of the allegation, the alleger's aluminum corrosion rates (Reference 2) were used as the input to Sandia's analysis. In Attachment 15 to Reference 1, the licensee converted the alleger's corrosion rates into the proper units for the COGAP input.

2.3 Corrosion of zinc

In the Palisades plant zinc can be found on galvanized steel surfaces and in paints. Similar to aluminum, the rates of corrosion of zinc depends on temperature and pH values.

a) Corrosion rate of zinc on galvanized surfaces

The data used in the FSAR analysis are identical to the data in Reference 5. To evaluate the impact of the allegation, the staff chose to use the alleger's corrosion rates converted to the proper units (Attachment 15 to Reference 1) as input data to Sandia's analysis.

However, after Sandia's analysis was completed, the staff discovered an additional source of information (Figure 2) on corrosion of zinc on galvanized surfaces from an experimental program performed at Sandia National Laboratories (Reference 4). The staff obtained and evaluated the information and made an engineering judgment that the effect is insignificant to its conclusion. A reanalysis is not necessary since the time dependent hydrogen generation from zinc, as shown in Reference 2, is a small fraction of the total hydrogen generation.

b) Corrosion rate of zinc in paints

The information on corrosion of zinc paints comes from a test program performed at Sandia National Laboratories (Reference 7). The corrosion rates are a function of temperature and pH values. As shown in Figure 3, the corrosion rates from Reference 7 are significantly higher than the constant corrosion rate used in the FSAR, which is test data obtained by the licensee. The large difference is probably from the difference in test conditions; the tests reported in Reference 7 were run at conditions more severe than those in the licensee's test. The corrosion rates used by the alleger appear to be between FSAR and Reference 7. For the same reason discussed above (this is a relatively insignificant term), the corrosion rates of zinc in paints used in Sandia's analysis were taken from the licensee's review analysis (Attachment 15 to Reference 1), which were the same values used for the corrosion rates for zinc on galvanized surfaces.

2.4 The effect of partial pressures and temperature in containment atmosphere and the effect of pressure and temperature at the recombiner intake

The original FSAR analysis was performed in 1976, and the method of analysis is probably outdated. However, the temperature and pressure effects are included in the COGAP code. The licensee has been using the COGAP code to perform its review analysis. Therefore, Sandia's analysis and the licensee's review analysis respond the alleger's concern in this area. However, the FSAR should be revised to incorporate these effects.

2.5 Correction of free volume from effective spray volume

The alleger assumed 95 percent of the free volume to be effective for hydrogen mixing because the containment spray covered only 90 percent of the free volume. The staff does not believe that this assumption is valid. The total free volume should be used because the turbulent mixing generated by the break flow jets, containment sprays, and natural convection flows will assure the hydrogen is well mixed within the entire containment free volume, including that outside of the spray volume.

2.6 Initiation time of hydrogen recombiner

The alleger stated that "Most important is the fact that the Palisades FSAR Section 14.22 allows plant to reach the control limit of 3.6 v/o hydrogen before initiation of recombiner operation. Because the recombiner by its nature of small hydrogen intake does not reduce immediately the hydrogen content of the containment atmosphere, its action is not sufficient to bring the hydrogen concentration down immediately, and in fact the concentration would be still rising to the level much higher than 4.1 % flammability limit because of the continuous production of hydrogen from all the other sources. This obviously could be catastrophic." Furthermore, the alleger infers that initiation of the recombiners might not occur until approximately 800 hours following the LOCA.

The staff reviewed the statement in the FSAR and got a somewhat different impression that the recombiner could initiate as late as the time when hydrogen concentration reached the limit of 3.6 v/o, not at 800 hours. Additionally, the staff reviewed the Palisades Emergency Operating Procedures (EOPs) (Attachment 17 to Reference 1). It is stated in the EOPs that at least one hydrogen recombiner should be in operation if containment pressure exceeds 3.7 psig. In a large LOCA the containment pressure would reach 3.7 psig in a few seconds. Additionally, the procedures require operators to perform checks of safety system status using the Safety Function Status Check Sheet on a continuous basis at approximately 15 minutes intervals. The licensee's review analysis

(Attachment 15 to Reference 1) assumed recombiner initiation time of either 2 hours or 24 hours. In either case, the calculated maximum hydrogen concentration was less than 4 v/o. The staff believes 2 hours is a proper assumption and chose it as the input for Sandia's analysis.

Based on our review of the Palisades FSAR and EOPs, the staff determined that the EOPs provide assurance that the hydrogen concentration will not exceed the flammable limit of 4 v/o hydrogen following a LOCA. Furthermore 4 v/o hydrogen is a lower flammable limit; it is not the limit to reach a catastrophic detonation. However, the FSAR should be revised to be consistent with the EOPs for the initiation time of recombiners. Otherwise, the FSAR could mislead the operators and could introduce undesirable risks.

2.7 Improper solution of the differential equation

The alleger indicated that the differential equation governing the production and consumption of hydrogen following recombiner initiation was not solved properly in the FSAR analysis. Similar to the above discussion in Section 2.4, the COGAP code solves the equation properly. Therefore, the Sandia analysis and the licensee's review analysis respond to the alleger's concern in this area. The FSAR should be revised to reflect the proper solution of the differential equations.

3.0 COGAP Calculation

The licensee has performed a COGAP calculation using the alleger's numbers from Reference 2. However, in reviewing the licensee's COGAP calculation and the alleger's analysis, the staff identified several input data that are not consistent with RG 1.7. Our consultant at Sandia National Laboratories performed COGAP calculations with parametric studies and found that the impact from those deviations could be significant. Initially, Sandia's analysis confirmed the licensee's results by using the same set of input as shown in Attachment 15 to Reference 1. Finally, Sandia made pertinent input changes that will be discussed in the following sections.

Percent of Zirconium Reacted: A value of 2.0 % was used by the alleger and the FSAR. However, a value of 1.0 % was used by Sandia based on Table 1 of RG 1.7, which states that "hydrogen production is 5 times the extent of the maximum calculated reaction under 10 CFR Part 50, Section 50.46, or that amount that would be evolved from a core-wide average depth of reaction into the original cladding of 0.00023 inch, whichever is greater." Based on Section 14.22.2 of the FSAR the former number is 5 times 0.2 % (i.e. 1%), and based on Attachment 14 to Reference 1 the later number is 0.84 % (less than 1%). Therefore, a value of 1.0 % is used in Sandia's analysis.

Radiolytic Hydrogen Yield: It was not clear what the basis was for the value of 0.3 molecules/100 eV used in the licensee's COGAP analysis. Based on Table 1 of RG 1.7, a value of 0.5 is used in Sandia's analysis.

Containment Volume: The value of $1.446*10^6$ ft^3 was used by the alleger because it was alleged that only the effective volume of the spray should be used. As discussed in Section 2.5, the entire free volume of $1.64*10^6$ ft^3 is used in Sandia's analysis.

Zinc in paint: In Attachment 15 to Reference 1, the licensee lumped the hydrogen generation from galvanized steel and from paint together. The area and the corrosion rate of the galvanized steel were used. In Sandia's analysis, the amount of zinc in paint specified in Table 14.22-4 of the FSAR (11,500 lb with a thickness of 0.003 inch) was used. This may conservatively generate a higher amount of hydrogen than either the licensee's results or the alleger's results. The licensee indicated that the amount of zinc in paint might be only half of the FSAR value. However, the staff continued to use the FSAR value.

Fission-Product Decay Energy Absorbed (FPDEA): This is an important consideration because it affects the amount of water that is dissociated into H_2 and O_2. The RC(9) and RC(10) designation was used in the COGAP manual (Reference 8).

RC(9) - This is the percent FPDEA absorbed by the core water. The value used by the licensee in Attachment 15 to Reference 1 was 7.0%. Based on RG 1.7, a value of 10.0% was used in the Sandia analysis.

RC(10) - This is the percent of total solid FPDEA absorbed by the sump water. The value used by the licensee was 12.0%. The licensee explained that the 12% value used by the alleger corresponded to severe accident conditions. Based on RG 1.7, a value of 1.0% was used in the Sandia's analysis which corresponds to a design basis accident.

3.1 Calculation Results

Sandia performed COGAP calculations which incorporated all of the alleger's concerns evaluated in Sections 2.1 through 2.7 and the pertinent changes to the licensee's COGAP calculation discussed in Section 3.0. The maximum hydrogen concentration obtained from Sandia's calculation was 2.9 v/o at 8 days.

4.0 CONCLUSION

The staff determined that portions of the allegation have some merit. The staff's consultant at Sandia Laboratories performed an analysis to evaluate the safety significance of the allegation. Based on the results of Sandia's analysis, the calculated hydrogen concentration following a LOCA at Palisades is 2.9 v/o, which does not exceed the 4.0 v/o flammable limit specified in RG 1.7. Therefore, the impact of the allegations is not as serious as the alleger described. However, the maximum hydrogen concentration from the original FSAR analysis was 2.0 v/o. Therefore, the margin of safety documented in the FSAR has been reduced. The licensee should revise the FSAR to reflect the new data, such as, amount of alumnium insulation, amount of zinc paint, new corrosion rates, recombiner initiation time, new method of analysis using COGAP code, input parameters, and the resulting hydrogen concentration.

5.0 REFERENCES

1. Memorandum from H. J. Miller to Dennis M. Crutchfield, dated September 20, 1990, Subject: Request for Assistance Allegations Regarding the Palisades Nuclear Plant.

Attachment 15: Licensee's Review and Analysis Package on the Alleger's Hydrogen Generation Analysis, dated February 5, 1990.

Attachment 17: Emergency Operating Procedures at Palisades Nuclear Plant, Revision 2, Dated July 20, 1990.

2. Alleger's Hydrogen Analysis File, SA-88-28, Revision 1, dated May 23, 1989, Subject: H2 Generation Following LOCA.

3. ORNL-3541, "Effect of Heat Flux on the Corrosion of Aluminum by Water," by J. C. Griess, H. C. Savage, and J. L. English, February 1964.

4. NUREG/CR-3361, "The Effect of Water Chemistry on the Rates of Hydrogen Generation from Galvanized Steel Corrosion at Post-LOCA Conditions," by V. M. Loyola and J. E. Womelsduff, December 1984.

5. ORNL-TM-2412, "Design Consideration of Reactor Containment Spray Systems. Part III, "The Corrosion of Materials in Spray Solutions," by J. C. Griess and A. L. Baccarella, December 1969.

6. "The Corrosion Handbook," by H. H. Uhlig, Eleventh Printing, November 1969.

7. NUREG/CR-3808, "The Effect of Post-LOCA Conditions on a Protective Coating (Paint) for the Nuclear Power Industry," by V. M. Loyola and J. E. Womelsduff, March 1985.

8. NUREG/CR-2847, "COGAP: A Nuclear Power Plant Containment Hydrogen Control System Evaluation Code," by R. G. Gido, January 1983.

9. Memorandum from C. Y. Cheng to Conrad E. McCracken, dated December 27, 1990, Subject: Corrosion of Aluminum and Zinc in Palisades Plant.

Principal Contributor: C. Li

Amount of Aluminum Corroded
(Comparison of FSAR Data with the Amounts
Calculated using Time Dependent Corrosion Rates)

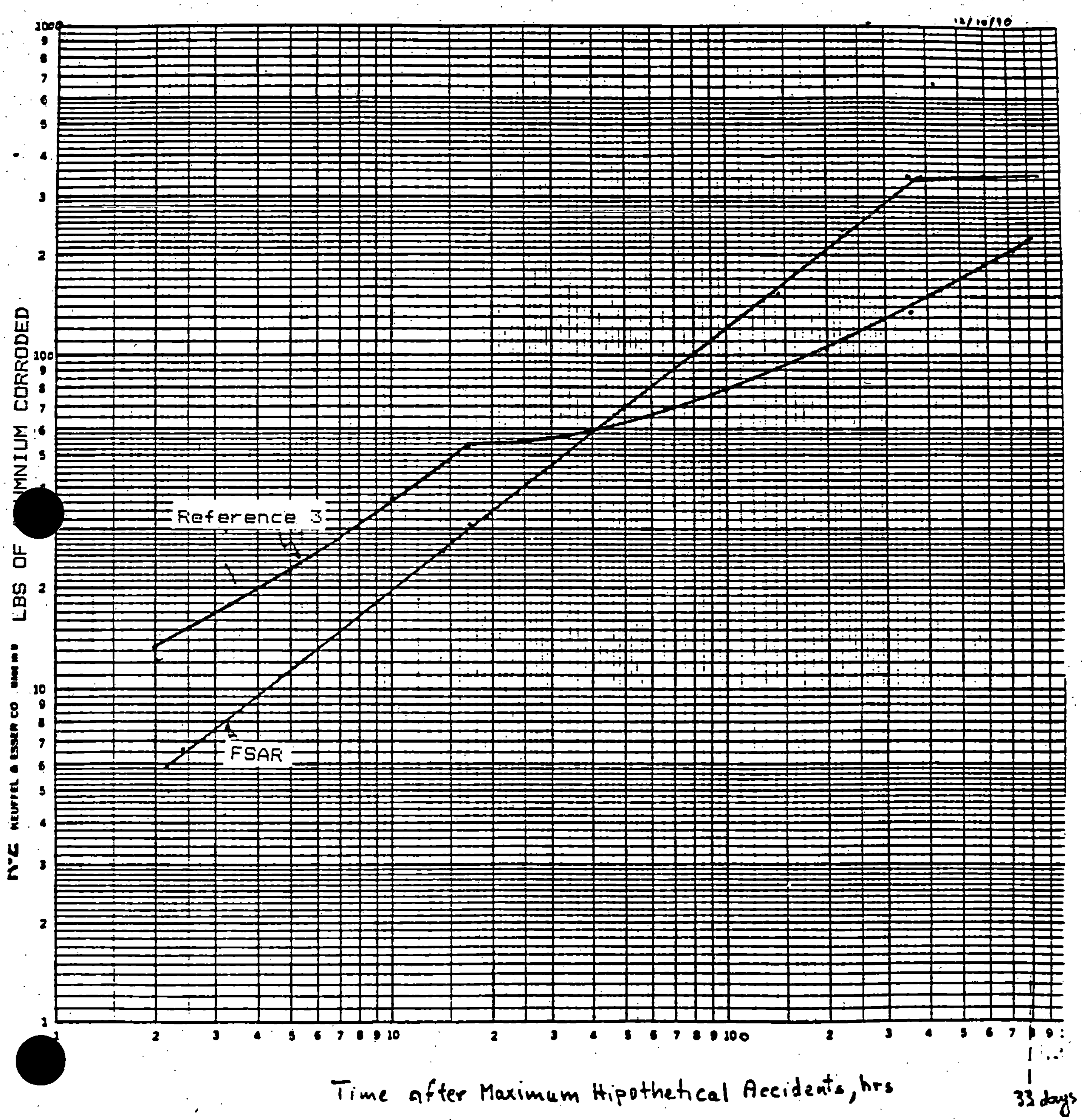

FiG. 1

Corrosion of Zinc on Galvanized Steel in Post-Accident Environment

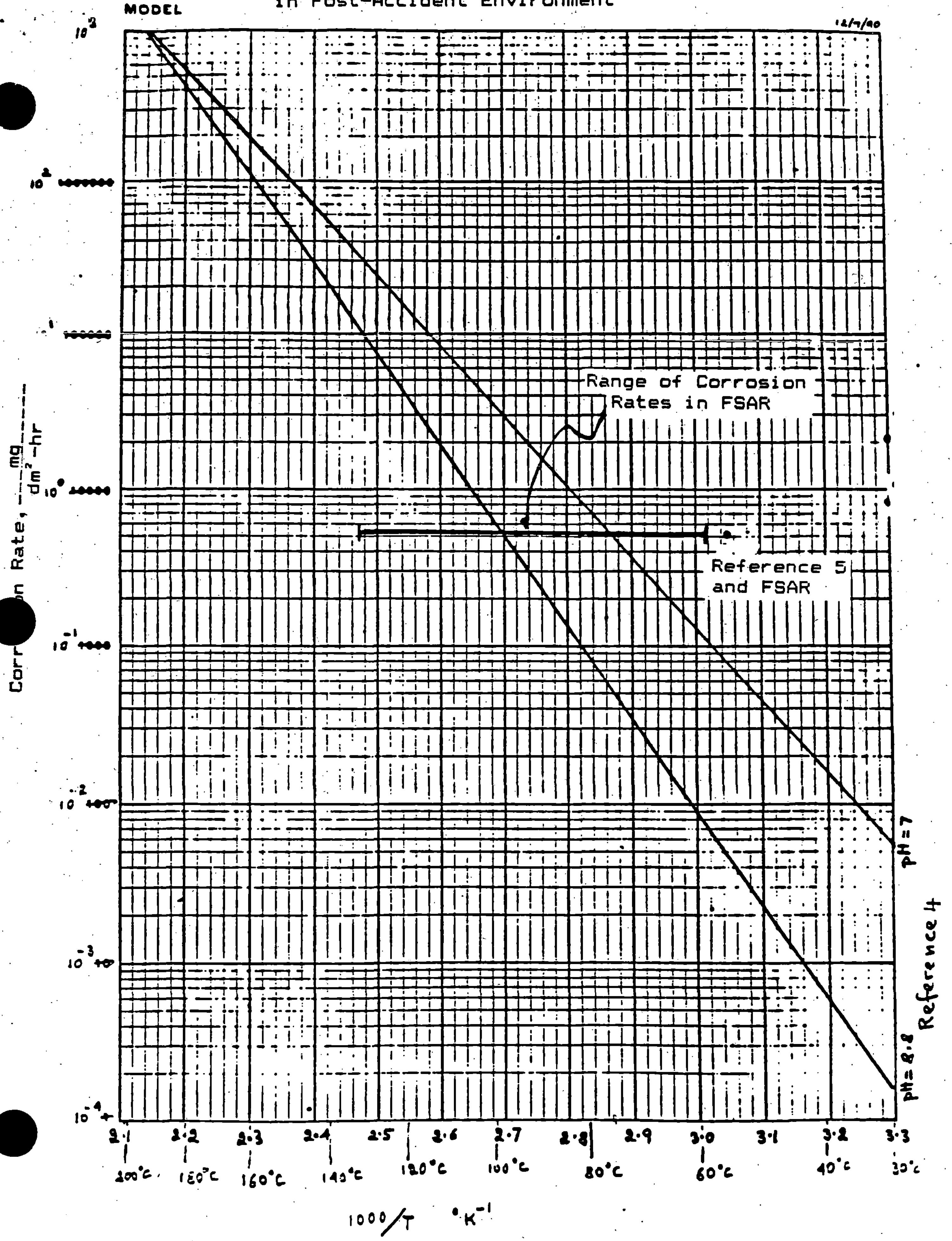

Fig 2

Corrosion of Zinc in Paint in Post-Accident Environment
(Primer+Topcoat)

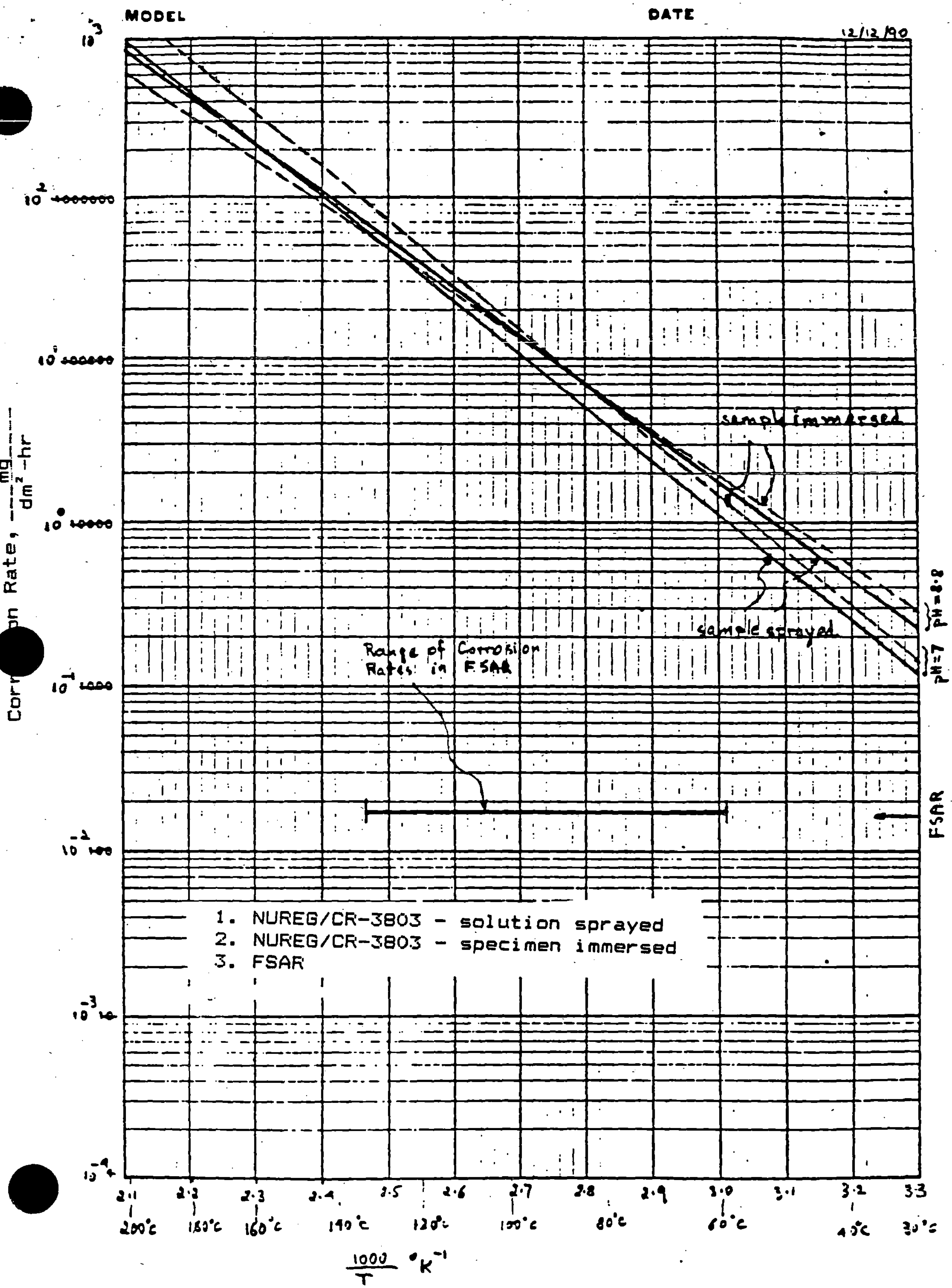

FIG. 3

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 136 TO PROVISIONAL OPERATING LICENSE NO. DPR-20

CONSUMERS POWER COMPANY

PALISADES PLANT

DOCKET NO. 50-255

1.0 INTRODUCTION

CHANGE NO. 1

By letter dated November 2, 1990, Consumers Power Company (the licensee) requested amendment to the Technical Specifications (TSs) appended to Provisional Operating License No. DPR-20 for the Palisades Plant. The proposed amendment would allow use of the Regulatory Guide 1.97 qualified neutron monitoring system which is being installed during the 1990 refueling outage. Additionally, a change was proposed to the Design Features section to more accurately describe the fixed absorber rods.

CHANGE NO. 2

By letter dated June 13, 1990, and subsequently revised by letters dated November 9 and December 7, 1990, and January 24, 1991, the licensee requested an amendment to revise TS 3.3.1.b., "Emergency Core Cooling System." The proposed amendment would reduce the required minimum boron solution level in the Safety Injection Tanks (SIT) from 186 to 174 inches. Additionally, the maximum allowed tank level would be expanded from 198 to 200 inches. This change effectively broadens the operating band at which SIT level must be maintained from 12 to 26 inches.

Two related TS changes were also submitted. First, a new surveillance requirement to check the SIT high and low level alarms was proposed to be included in TS table 4.1.2. Secondly, the Bases section for TS 3.3.1 has been updated and two TS references have been added.

2.0 DISCUSSION

CHANGE NO. 1

In 1988, the licensee performed a modification which upgraded the sensitivity of the fission chambers used to detect neutron flux. During the most recent refueling outage, the licensee has completed its upgrade of the neutron monitoring system by certifying the system is qualified to the criteria of Regulatory Guide 1.97. The changes in the neutron monitoring system performed this outage involved using the existing fission chambers, installing new cables from the fission chambers through two new electric penetrations to preamplifiers (which were relocated to outside containment), and installing new cables from the preamplifiers to the power sources in the control room. Additionally, the neutron monitoring channel which supplies the alternate shutdown panel was modified. The alternate shutdown panel had previously received neutron monitoring indication from a dedicated, spare fission chamber. The new system supplies the panel through an optical isolator associated with the left channel (NI-1/3) of the RG 1.97 qualified instrumentation.

Eight channels of instrumentation are provided to monitor the neutron flux. The nuclear instrumentation system consists of two start-up channels, two wide-range logarithmic channels and four power range safety channels. The start-up and wide-range channels share high sensitivity fission chambers while the power ranges channels are completely independent (each power range channels has a separate detector and power supply).

The rate-of-change of power is normally monitored at start-up by two source range monitors which sum inputs from two fission chambers and cover a range of approximately five decades (control room indication uses a scale from 1 to 3x10e5 cps). The two other channels are wide-range units which take signals from fission chambers and cover a range greater than ten decades, overlapping the start-up channels by approximately three decades (control room indication uses a scale from 1x10e-8 to 200% power).

The proposed Technical Specification changes, associated with the above modifications are as follows:

<u>Changes</u>

A. In the third paragraph from the end of the Basis for Section 3.17, delete reference to the "start-up" range and replace it with "source" range. Delete reference to the "log" range and replace it with reference to the "wide" range.

B. In Table 3.17.1, change Item No. 3 from "Log Range" to "Wide-Range".

C. In Table 3.17.4, Item 7, change "Start-up" to "Source Range" and in footnote (d) to Table 3.17.4, change "log range" to "wide range".

D. In Table 3.25.1, Item No. 7, change "Start-up" to "Source" and "(N-001A)" to "(N-1/3C)".

E. In Table 3.25.1, Item 14, add "Neutron Monitor System Power" under the Function Column.

F. In Table 4.1.1, Item 2, under the Channel Description Column, delete the word "Logarithmic". In Surveillance Functional Column, add "C. Calibrate". In the Frequency Column, add "R" on the

"C. Calibrate" line. In the Surveillance Method Column on the "a" line, delete "both wide-range readings" and insert "Channel indications". Also in the Surveillance Method column, add "C. Channel alignment through measurement/adjustment of internal test points".

G. In Table 4.1.3, Item No. 1 in the Channel Description Column change "Start-up" to "Source"; in the Surveillance Functions Column add "c. Calibrate"; in the Frequency Column, add "R" on the "c. Calibrate" line; and, in the Surveillance Method Column, add "C. Channel alignment through measurement/adjustment of internal test points.

H. In Table 4.21.1, item No. 7, under the Channel Description column, change "Start-up" to "Source" and "(NI-001A)" to "(NI-1/3C)". In the surveillance Method Column, after "a. Internal Test Signal" add, "(Performed under Table 4.1.3, Item 1.b)".

I. In Section 5.3.2d, change the description of the mechanically fixed rods from "...mechanically fixed boron rods..." to"...mechanically fixed absorbers rods...".

CHANGE NO. 2

The four safety injection tanks are part of the safety injection system and are used to flood the core with borated water following a depressurization of the primary coolant system. Three of the four tanks will provide sufficient coolant to recover the core following a Loss of Coolant Accident. The tanks are connected to the Primary Coolant System cold legs through normally open isolation valves. Two check valves prevent primary coolant from entering the tanks. Current TS maintain the tanks pressurized to at least 200 psig, with a tank liquid level of at least 186 inches and a maximum level of 198 inches, and a boron concentration from 1720 to 2000 ppm. Injection will occur whenever the primary system pressure falls below the combined pressure of the static water head plus the tank gas pressure.

The licensee proposed the following changes to the TS;

<u>Changes</u>

A. Change Specification 3.3.1.b to read as follows:
"All four Safety Injection Tanks are operable and pressurized to at least 200 psig with a tank liquid level of at least 174 inches and maximum level of 200 inches with a boron concentration of at least 1720 ppm but not more than 2000 ppm."

B. Change the fourth paragraph of Section 3.3 Basis as follows:
"The limits for the Safety Injection Tank pressure and volume assure the required amount of water injection during an accident and are based on values used for the accident analyses (3, 4). The minimum 174-inch level corresponds to a volume of 1040 ft^3 and the maximum 200-inch level corresponds to a volume of 1176 ft^3."

C. Add the following to References:

"(3) FSAR, Section 14.17
(4) Letter, H. G. Shaw (ANF) to R. J. Gerling (CPCo), "Standard Review Plan Chapter 15, Disposition of Events Review for Changes to Technical Specification Limits on Palisades Safety Injection Tank Liquid Levels", April 11, 1990.

D. Add Surveillance Function "C." To Item 13 on Table 4.1.2 to require performance, at least once per 18 months, of a functional check on the SIT high and low level alarms.

These changes are considered necessary to reduce the risk of TS violations made possible by periodic surveillance and correction of boron concentration. When sampling the SITs to verify boron concentration, it is necessary to drain the tanks sufficiently to obtain an accurate sample. During this evolution, TS Section 3.3.2.a is in effect and limits the non-operability of one tank to one hour. Because a significant amount of water must be drained from the tank to obtain a representative sample, the possibility exists that proper level may not be restored within the one hour period. This procedure places demands on the Operations staff which would be minimized if the operating band of the tanks were broadened.

3.0 EVALUATION

CHANGE No. 1

The changes to the neutron monitoring system enhance the reliability of the accuracy of the neutron monitoring function under accident conditions. The previous system has basically been upgraded to the more stringent requirements of Regulatory Guide 1.97. The upgraded equipment performs the same function as the previously installed equipment, and maintains the same degree of redundancy.

For consistency with Standard Technical Specifications, the licensee has proposed to change the name of the most sensitive neutron monitoring range from "Start-up Range" to "Source-Range". Similarly, the name of the "Logarithmic Range" has been proposed to be renamed "Wide-Range". These designations are more consistent with standard industry phraseology, and more clearly describe the respective neutron monitoring ranges. Changes A,B,C,D,F,G and H reflect the new terminology.

Changes D and H also correct the designation of the source range neutron monitor which provides indication for alternate shutdown capability (N-001A is changed to NI-1/3C). The source of the signal to the alternate shutdown panel source range monitor has been modified. The previous signal (N-001A) originated from an older, dedicated fission chamber. The new signal (NI-1/3C) is supplied through an optical isolator associated with the left channel of the new, RG 1.97 qualified instrumentation. The newer system provides for more accurate indication and improved reliability.

Change E provides additional information to the TS, to clearly delineate the neutron monitoring system power source. Changes F, G, and H either add additional surveillances to the TS, or clarify existing surveillances.

Change I is necessary to correct the TS description of the core's mechanically fixed rods. Fixed boron rods are no longer in use at Palisades. A conversion from boron to Gadolinia rods was completed in core reloads H, I and J. Also, with recent fuel cycles converting to reduced leakage designs, additional neutron absorbent rods are being utilized (e.g., stainless steel, Hafnium). Because a variety of materials may be used, the term "fixed absorber rods" is considered appropriate to accommodate different core reload designs.

In summary, the proposed TS changes more clearly describe the function and operation of the neutron monitoring system. Additionally, the changed description of the type of neutron absorbing rod in use at the Palisades Plant corrects an error in the Design Features TS section, which was overlooked at the time of the Core reload H submittal. These changes reflect the use of material or equipment which will perform the same functions as existing equipment, and are considered acceptable.

CHANGE NO. 2

Discussions were held with the licensee's Operations staff to assess the operational restrictions imposed by the current SIT level band. Operational data for the period from 11/88 to 9/90 indicates that the SITs were sampled 187 times. SIT sampling is required monthly by the TS. The increased sampling frequency (approximately 100 samples above the TS minimum) was required due to known inleakage into the SITs from the primary coolant system. This inleakage slowly lowered SIT boron concentration; therefore, additional monitoring of tank concentration was required in order to ensure the minimum TS levels for SIT boron concentration were maintained.

The SITs do not have tank recirculation capability; therefore, a relatively large amount of tank water is required by procedure to be purged through the sample lines in order to obtain a representative sample (on the order of 1700 gallons per sample). This water volume necessitates that the TS Limiting Condition for Operation be entered each time a tank is sampled.

Although the most appropriate deterrent in preventing unnecessary SIT sampling is ensuring the leak-tightness of the SIT boundary valves, widening the operational water level band will assist in maintaining the tanks within their prescribed limits.

The licensee contracted with their fuel vendor to evaluate the effect of a slightly reduced or increased total liquid volume in the SITs (174 and 202 inches, respectively, for minimum and maximum allowed SIT levels). The result of the fuel vendor's Standard Review Plan Chapter 15 disposition of events was that the large break loss of coolant transient is the only event which completely drains the SITs (thereby it is the only event which could be affected by the changes in SIT level limits).

Data supplied by the licensee indicates that flow from the intact loop SITs, SIT lines, and cold legs keeps the downcomer full for about 30 seconds after the peak cladding temperature (PCT) for the transient is reached. Reduction of the minimum SIT level to 174 inches causes the SITs to empty, in the worst case, approximately four seconds earlier than would have occurred with the previous tank limits. Downcomer level does not fall prior to the time the PCT is reached. Additionally, increasing the maximum SIT level to 202 inches has no impact on the large break LOCA analysis because the SIT flow time would be conservatively extended beyond the time in the limiting analysis. These conclusions apply to all break sizes contained in the February 1990 Palisades Large Break LOCA Analysis of record.

The following factors were also considered:

- The change in the upper SIT limit involves a relatively small increase in the maximum amount of water stored in the SITs (two inches of level); therefore, the probability of overfilling, containment flooding, and malfunctions due to seismic events are not significantly increased.

- The LOCA containment analysis, which conservatively does not take credit for SIT injection, shows that peak containment pressure stays below the design pressure.

- A comparison of the SIT operating band volumes in use at several Combustion Engineering plants indicates similar volumes. Also, the Combustion Engineering Standard TS provides for an operating band of roughly the same volume as that proposed by Palisades (126 and 136 cubic feet, respectively).

- The effect of the reduced minimum SIT inventory on the available suction source for Safety Injection during long term recirculation from the containment sump has been considered. The reduction (1885 gal) in required minimum inventory is a very small fraction of the total available inventory (approx. 380,000 gal) considering the vast inventory contribution from the four SITs and the Safety Injection Refueling Water Tank; and, therefore, will have a negligible effect on the operation of the safety injection pumps or the containment temperature and pressure response.

- The boron concentration in the tanks will be unchanged and the slight reduction in total inventory will not have a significant effect on the sump boron concentration during the recirculation phase of the accident. Additionally, this change will not significantly effect the time before hot leg injection is required to prevent precipitation of boron.

In summary, the proposed changes to the Technical Specification limits on Palisades SIT levels have been evaluated to ensure that adequate water is available for make-up to the primary coolant system. The analysis shows that when the contents of the SITs are at the proposed lower level, and a large break LOCA occurs, the SITs do not empty until after the peak cladding temperature is reached and until after high and low pressure safety injection are actuated. The addition of a surveillance requirement to perform a functional check on the SIT high and low level alarms institutes additional TS controls on ensuring that SIT level will be adequately measured and maintained. Additionally, the basis section has been updated and the TS Reference section appropriately expanded. Therefore, these proposed TS changes are considered acceptable.

4.0 ENVIRONMENTAL CONSIDERATION

This amendment involves a change in a requirement with respect to the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and a change in a surveillance requirement. The staff has determined that the amendment involves 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 this amendment involves no significant hazards consideration and there has been no public comment on such finding. Accordingly, this amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR Section 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 this amendment.

5.0 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, and (2) such activities will be conducted in compliance with the Commission's regulations, and the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public. The staff therefore concludes that the proposed changes are acceptable.

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