TECHNICAL EVALUATION REPORT

PWR MAIN STEAM LINE BREAK WITH CONTINUED FEEDWATER ADDITION

PALISADES PLANT
CONSUMERS POWER COMPANY

NRC DOCKET NO. 50-255
NRC TAC NO. 46850
NRC CONTRACT NO. NRC-03-81-130

FRC PROJECT C5506
FRC ASSIGNMENT 5
FRC TASK 135

Prepared by

Franklin Research Center
20th and Race Street
Philadelphia, PA 19103

Author: F. W. Vosbury

FRC Group Leader: R. C. Herrick

Prepared for

Nuclear Regulatory Commission
Washington, D.C. 20555

Lead NRC Engineer: P. Hearn

November 29, 1983

Prepared by:	Reviewed by:	Approved by:
F W Vosbury	R C Herrick	Vu Ngoc Con
Principal Author	Group Leader	Department Director (Acting)
Date: 11.29.83	Date: 11/29/83	Date: 11-29-83

Franklin Research Center
A Division of The Franklin Institute
The Benjamin Franklin Parkway, Phila., Pa. 19103 (215) 448-1000

CONTENTS

FOREWORD

This Technical Evaluation Report was prepared by Franklin Research Center under a contract with the U.S. Nuclear Regulatory Commission (Office of Nuclear Reactor Regulation, Division of Operating Reactors) for technical assistance in support of NRC operating reactor licensing actions. The technical evaluation was conducted in accordance with criteria established by the NRC.

Mr. F. W. Vosbury contributed to the technical preparation of this report through a subcontract with WESTEC Services, Inc.

1. INTRODUCTION

1.1 PURPOSE OF REVIEW

This Technical Evaluation Report (TER) documents an independent review of the Consumers Power Company's (CPC) compliance with the Nuclear Regulatory Commission's (NRC) IE Bulletin 80-04, "Analysis of a Pressurized Water Reactor Main Steam Line Break with Continued Feedwater Addition" [1], as it pertains to the Palisades Plant. This evaluation was performed with the following objectives:

- o to assess the conformance of CPC's main steam line break (MSLB) analyses with the requirements of IE Bulletin 80-04
- o to assess CPC's proposed interim and long-range corrective action plans and schedules if needed as a result of the MSLB analyses.

1.2 GENERIC BACKGROUND

In the summer of 1979, a pressurized water reactor (PWR) licensee submitted a report to the NRC that identified a deficiency in the plant's original analysis of the containment pressurization resulting from a MSLB. A reanalysis of the containment pressure response following a MSLB was performed, and it was determined that, if the auxiliary feedwater (AFW) system continued to supply feedwater at runout conditions to the steam generator that had experienced the steam line break, containment design pressure would be exceeded in approximately 10 minutes. The long-term blowdown of the water supplied by the AFW system had not been considered in the earlier analysis.

On October 1, 1979, the foregoing information was provided to all holders of operating licenses and construction permits as IE Information Notice 79-24 [2]. Another facility performed an accident analysis review pursuant to receipt of the information in the notice and discovered that, with offsite electrical power available, the condensate pumps would feed the affected steam generator at an excessive rate. This excessive feed was not previously considered in the plant's analysis of a MSLB accident.

A third licensee informed the NRC of an error in the MSLB analysis for their plant. During a review of the MSLB analysis, for zero or low power at the end of core life, the licensee identified an incorrect postulation that the startup feedwater control valves would remain positioned "as is" during the transient. In reality, the startup feedwater control valves will ramp to 80% full open due to an override signal resulting from the low steam generator pressure reactor trip signal. Reanalysis of the events showed that opening of the startup valve and associated high feedwater addition to the affected steam generator would cause a rapid reactor cooldown and resultant reactor return-to-power response, a condition which is outside the plant design basis.

Because of these deficiencies identified in original MSLB accident analyses, the NRC issued IE Bulletin 80-04 on February 8, 1980. This bulletin required all PWRs with operating licenses and certain near-term PWR operating license applicants to perform the following:

"1. Review the containment pressure response analysis to determine if the potential for containment overpressure for a main steam line break inside containment included the impact of runout flow from the auxiliary feedwater system and the impact of other energy sources, such as continuation of feedwater or condensate flow. In your review, consider your ability to detect and isolate the damaged steam generator from these sources and the ability of the pumps to remain operable after extended operation at runout flow.

2. Review your analysis of the reactivity increase which results from a main steam line break inside or outside containment. This review should consider the reactor cooldown rate and the potential for the reactor to return to power with the most reactive control rod in the fully withdrawn position. If your previous analysis did not consider all potential water sources (such as those listed in 1 above) and if the reactivity increase is greater than previous analysis indicated the report of this review should include:

    a. The boundary conditions for the analysis, e.g., the end of life shutdown margin, the moderator temperature coefficient, power level and the net effect of the associated steam generator water inventory on the reactor system cooling, etc.,

    b. The most restrictive single active failure in the safety injection system and the effect of that failure on delaying the delivery of high concentration boric acid solution to the reactor coolant system,

c. The effect of extended water supply to the affected steam generator on the core criticality and return to power,

d. The hot channel factors corresponding to the most reactive rod in the fully withdrawn position at the end of life, and the Minimum Departure from Nucleate Boiling Ratio (MDNBR) values for the analyzed transient.

3. If the potential for containment overpressure exists or the reactor return-to-power response worsens, provide a proposed corrective action and a schedule for completion of the corrective action. If the unit is operating, provide a description of any interim action that will be taken until the proposed corrective action is completed."

1.3 PLANT-SPECIFIC BACKGROUND

CPC responded to IE Bulletin 80-04 in a letter to the NRC dated May 9, 1980 [3]. On March 8, 1982 [4] and September 7, 1982 [5], the NRC forwarded to the Licensee a request for additional information necessary for the completion of this report. CPC responded to these requests on April 26, 1982 [6], October 26, 1982 [7], July 22, 1983 [8], and September 8, 1983 [9]. Information in References 3, 6, 7, 8, and 9; the Palisades Plant Final Safety Analysis Report (FSAR) [10]; NUREG-0820, which reported on the effects of a MSLB [11]; and a letter to the NRC dated May 29, 1981 [12] describing the conceptual design of an improved AFW system at the Palisades Plant have been evaluated to determine the adequacy of the Licensee's compliance with IE Bulletin 80-04.

2. ACCEPTANCE CRITERIA

The following criteria against which the Licensee's MSLB response was evaluated were provided by the NRC [13]:

1. PWR licensees' responses to IE Bulletin 80-04 shall include the following information related to their analysis of containment pressure and core reactivity response to a MSLB within or outside containment:

   a. A discussion of the continuation of flow to the affected steam generator, including the impact of runout flow from the AFW system and the impact of other energy sources, such as continuation of feedwater or condensate flow. AFW system runout flow should be determined from the manufacturer's pump curves at no backpressure, unless the system contains reliable anti-runout provisions or a more representative backpressure has been conservatively calculated. If a licensee assumes credit for anti-runout provisions, then justification and/or documentation used to determine that the provisions are reliable should be provided. Examples of devices for which provisions are reliable are anti-runout devices that use active components (e.g., automatically throttled valves) which meet the requirements of IEEE Std 279-1971 [14] and passive devices (e.g., flow orifices or cavitating venturis).

   b. A determination of potential containment overpressure as a result of the impact of runout flow from the AFW system or the impact of other energy sources such as continuation of feedwater or condensate flow. Where a revised analysis is submitted or where reference is made to the existing FSAR analysis, the analysis must show that runout AFW flow was included and that design containment pressure was not exceeded.

   c. A discussion of the ability to detect and isolate the damaged steam generator from continued feedwater addition during the MSLB accident. Operator action to isolate AFW flow to the affected steam generator within the first 30 minutes of the start of the MSLB should be justified. If operator action is to be completed within the first 10 minutes, then the justification should address the indication available to the operator and the actions required. Where operator action is required to prevent exceeding a design value, i.e., containment design pressure or specified acceptable fuel design limits, then the discussion should include the calculated time when the design value would be exceeded if no operator action were assumed. Where operator actions are to be performed between 10 and 30 minutes after the start of the MSLB, the justification should address the indications available to the operator and the operator actions required, noting that for the first 30 minutes, all actions should be performed from the control room.

d. Where all water sources were not considered in the previous analysis, an indication should be provided of the core reactivity change which results from the inclusion of additional water sources. A submittal which does not determine the magnitude of reactivity change from an original analysis is not responsive to the requirements of IE Bulletin 80-04.

2. If containment overpressure or a worsening of the reactor return-to-power with a violation of the specified acceptable fuel design limits described in Section 4.2 of the Standard Review Plan [15] (i.e., increase in core reactivity) can occur by the licensee's analysis, the licensee shall provide the following additional information:

   a. the proposed corrective actions to prevent containment overpressure or the violation of fuel design limits and the schedule for their completion

   b. the interim actions that will be taken until the proposed corrective action is completed, if the unit is operating.

3. The acceptable input assumptions used in the licensee's analysis of the core reactivity changes during a MSLB are given in Section 15.1.5 of the Standard Review Plan [16]. The following specific assumptions should be used unless the analysis shows that a different assumption is more limiting:

   Assumption II.3.b.: Analysis should be performed to determine the most conservative assumption with respect to a loss of electrical power. A reactivity analysis should be conducted for a normal power situation as well as a loss of offsite power scenario, unless the licensee has previously conducted a sensitivity analysis which demonstrates that a particular assumption is more conservative.

   Assumption II.3.d.: The most restrictive single active failure in the safety injection system which has the effect of delaying the delivery of high concentration boric acid solution to the reactor coolant system, or any other single active failure affecting the plant response, should be considered.

Assumption II.3.g.: The initial core flow should be chosen such that the post-MSLB shutdown margin is minimized (i.e., maximum initial core flow).

The acceptable computer codes for the licensee's analysis of core reactivity changes are, by nuclear steam supply system (NSSS) vendor, the following: CESEC (Combustion Engineering), LOFTRAN (Westinghouse), and TRAP (Babcock & Wilcox). Other computer codes may be used, provided that these codes have previously been reviewed and found to be acceptable by the NRC staff. If a computer code is used which has not been reviewed, the licensee must describe the method employed to verify the code results in sufficient detail to permit the code to be reviewed for acceptability.

4. If the AFW pumps can be damaged by extended operation at runout flow, the licensee's action to preclude damage should be reviewed for technical merit. Any active features should satisfy the requirements of IEEE Std 279-1971. Where no corrective action has been proposed, this should be indicated to the NRC for fu :her action and resolution.

5. The electrical instrumentation and controls needed to detect and initiate isolation of the affected steam generator and feedwater sources in order to prevent containment overpressure and/or unacceptable core reactivity increases must satisfy safety-grade requirements. Instrumentation that the operator relies upon to follow the accident and to determine isolation of the affected steam generator and feedwater sources should conform to the criteria contained in ANS/ANSI-4.5-1980, "Criteria for Accident Monitoring Functions in Light-Water-Cooled Reactors" [17], and the regulatory positions in Regulatory Guide 1.97, Rev. 2, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident" [18].

6. AFW system status should be reviewed to ensure that system heat removal capacity does not decrease below the minimum required level as a result of isolation of the affected steam generator and also that recent changes have not been made in the system which adversely affect vital assumptions of the containment pressure and core reactivity response analyses.

7. The safety-grade requirements (redundancy, seismic and environmental qualifications, etc.) of the valves that isolate the main feedwater (MFW) and AFW systems from the affected steam generator should be specified. Isolation valves that are relied upon to isolate the MFW and AFW systems from the affected steam generator should satisfy the following criteria to be considered safety-grade:

   o Redundancy and power source requirements: The isolation valves should be designed to accommodate a single failure. A failure-modes-and-effects analysis should demonstrate that the system is capable of withstanding a single failure without loss of function. The single failure analysis should be conducted in accordance with the appropriate rules of application of ANS-51.7/N658-1976, "Single Failure Criteria for PWR Fluid Systems" [19].

- Seismic requirements: The isolation valves should be designed to Category I as recommended in Regulatory Guide 1.26 [20].
- Environmental qualification: The isolation valves should satisfy the requirements of NUREG-0588, Rev. 1, "Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment" [21].
- Quality standards: The isolation valves should satisfy Group B quality standards as recommended in Regulatory Guide 1.26 or similar quality standards from the plant's licensing bases.

3. TECHNICAL EVALUATION

The scope of work under the NRC contract included the following:

1. Review the Licensee's response to IE Bulletin 80-04 against the acceptance criteria.

2. a. Evaluate the Licensee's MSLB analyses for the potential of overpressurizing the containment and with respect to the core reactivity increase due to the effect of continued feedwater flow

   b. Evaluate the Licensee's proposed corrective actions and schedule for implementation if the findings of Task 2a indicate that a potential exists for overpressurizing the containment or worsening the reactor return-to-power in the event of a MSLB accident.

3. Prepare a TER for each plant based on the evaluation of the information presented for Tasks 1 and 2 above.

This report constitutes a TER in satisfaction of Task 3. Sections 3.1 through 3.3 of this report state the requirements of IE Bulletin 80-04 by subsection, summarize the Licensee's statements and conclusions regarding these requirements, provide an evaluation of the Licensee's statements and conclusions, and present overall conclusions and recommendations.

3.1 REVIEW OF CONTAINMENT PRESSURE RESPONSE ANALYSIS

The requirement from IE Bulletin 80-04, Item 1, is as follows:

"Review the containment pressure response analysis to determine if the potential for containment overpressure for a main steam line break inside containment included the impact of runout flow from the auxiliary feedwater system and the impact of other energy sources, such as continuation of feedwater or condensate flow. In your review, consider your ability to detect and isolate the damaged steam generator from these sources and the ability of the pumps to remain operable after extended operation at runout flow."

3.1.1 Summary of Licensee Statements and Conclusions

In regard to the review of the containment pressure response analysis for the Palisades Plant, the Licensee stated [3]:

"Consumers Power Company has reviewed the containment pressure response analysis to determine if this analysis included the impact of runout flow from the auxiliary feedwater system as well as the impact of other energy sources (e.g., continuation of feedwater or condensate flow). As discussed in the bulletin, a design oversight regarding the delivery of condensate flow to a ruptured steam generator was previously identified. This deficiency is presently being addressed as described in Licensee Event Report 79-041.

The current review identified that the initial analysis did not include the impact of runout flow from the auxiliary feedwater system (AFWS). Auxiliary feedwater flow was probably not considered because the AFWS was originally intended to be a manually actuated system. This assumed that for the large steam line break, which might be affected adversely by auxiliary feedwater flow, the operator would recognize the intact steam generator and provide feedwater to that steam generator only. Recognition of the intact steam generator would rely on the large indicated pressure differential between the two steam generators.

A plant modification is presently being performed to provide automatic starting of the Palisades AFWS. An analysis has been conducted to evaluate the effect of auxiliary feedwater flow on the containment pressure response in a Main Steam Line Break (MSLB). The assumptions made in the analysis are as follows:

a. Double-ended guillotine rupture of a 36" main steam line at the steam generator nozzle.

b. Initial reactor power of 2650 MWt (licensed power is 2530 MWt). The full power case is more limiting than the zero power case because of the single failure assumption (Assumption C). For the zero power case, the worst single failure is loss of one of the three containment spray (CTS) pumps (loss of off-site power is not considered credible in the zero power case). With two CTS pumps operating, the cooldown of the containment will proceed more rapidly; thus, the full power case bounds the zero power case.

c. Loss of off-site power and failure of a diesel generator to start results in only one containment spray pump and three air coolers being available to cool the containment. The steam generator blowdown analysis assumed the availability of off-site power to run the primary coolant pumps and, thereby, maximize the rate and magnitude of the initial energy release to the containment.

d. Main feedwater flow rampdown from full flow at time of trip to zero flow at 60 seconds post-reactor trip.

e. Full containment spray flow from one pump at 60 seconds after an MSLB, and main steam line isolation 2 seconds after an MSLB. Spray flow time and main steam line isolation valve closure time are important with respect to the magnitude of the initial pressure peak. However, these have little or no effect on the long-term cooldown of the containment.

f. Runout flow from one auxiliary feedwater pump (assumed to be 750 gpm) is initiated at 120 seconds after the MSLB. The automatic AFWS actuation system incorporates a 2-minute timer delay and an automatic flow controller which will be administratively set to less than 250 gpm per steam generator. Failure of the automatic controller to provide maximum flow demand is assumed even though this represents a second single failure in addition to the diesel generator failure. The automatic AFWS logic blocks the start of the steam-driven auxiliary feedwater pump, unless the motor-driven pump does not start and deliver flow. Therefore, only one pump is assumed to be operating. (Ref: Consumers Power Company letter dated 4-23-80.)

For conservatism, the 750 gpm of cold feedwater was assumed to flash to steam with an enthalpy of 1183.1 Btu/lbm (saturated enthalpy at 80 psia). This assumption ignores any physical limitations on the amount of energy available within the primary coolant system to boil the cold feedwater.

The runout flow from one auxiliary feedwater pump will be less than 750 gpm (runout flow has been calculated conservatively to be less than 650 gpm). This calculation was based on a steam generator pressure of 0 psig and with known pump and injection valve characteristics."

Regarding the AFW pump's ability to remain operable after extended operation at runout flow, the Licensee stated:

"Acceptable pump performance at flow rates up to 690 gpm and with a net positive suction head (NPSH) of less than 10 feet (available NPSH exceeds 20 feet during actual operation) has been verified by vendor test."

In conclusion, the Licensee stated:

"The results of this analysis...[show a peak containment pressure of 68.5 psia occurring at 60 seconds.] A single CTS pump (and three air coolers) are found to be more than adequate in removing the energy being deposited in containment as a result of auxiliary feedwater addition. At 30 minutes, the containment pressure is continuing to slowly decrease. It is assumed that by 30 minutes, the operator would recognize (by observing the steam generator pressure difference) that he was feeding the broken steam generator, and terminate feedwater to it."

As part of this review, the NRC requested additional information about the effects of a single failure to the MFW regulating valves or MFW bypass valves causing either to remain open after receiving a low steam generator pressure trip signal. The failure of these valves would allow the condensate pumps to continue pumping water through the MFW pumps into the ruptured steam generator. In response to this request, the Licensee stated [7]:

"If offsite power remains available following a Main Steam Line Break (MSLB) the condensate pumps will continue to run. Since the ruptured steam generator depressurizes to the containment pressure the condensate pumps will provide runout flow to the steam generator. To account for this the operators of the feedwater regulating valves have been modified to close on steam generator low pressure. This prohibits any additional feedwater to the steam generator over and above that assumed in the MSLB analyses. These analyses, however, assumed what were considered the worst single failures. The failure of a feedwater regulating valve to close was not analyzed.

The containment response to a MSLB was performed by Consumers Power Company in response to LER-80-003. That analysis assumed loss of offsite power along with failure of diesel/generator 1-1 which results in only one containment spray pump and three containment air coolers. This was considered to be the worst case for two reasons:

1. Delay in containment spray due to both loss of offsite power and the increased time to fill the spray headers with one spray pump

2. Decreased heat removal capability with only one spray pump and the decreased effectiveness of containment air coolers in superheated atmosphere.

The containment response to a MSLB with offsite power available but with a failure of feedwater regulating valve to close has not been analyzed. It is felt that this scenario is not as severe as that assumed in the above analyses because of immediate containment spray and the availability of three containment spray pumps and four containment air coolers. However, work is in progress to calculate the containment response explicitly. The key to this analysis is the flow into the steam generator due to the condensate pumps. Since the condensate pumps will be pumping through the ramping down feedwater pumps the total flow is not easily determined. However, this calculation can be performed by RETRAN. Presently, a model of the condensate/feedwater train is being developed for RETRAN input. It is expected that this analysis along with the resulting containment response analysis will be complete by the end of the year. Upon completion of the analysis a submittal will be forthcoming to the NRC."

The Licensee submitted the following analysis in Reference 8.

"Consumers Power Company has completed the analysis and determined that meeting the new NRC-imposed system requirement will require a modification to the main feedwater system. This modification which is presently scheduled to be completed during 1985 Refueling Outage involves modifying the control circuitry of the main feedwater stop valves, CV-0742 and -0744 and feedwater bypass valves CV-0734 and -0735 to close on low steam generator pressure. Also included in this modification will be the addition of a control valve in series with CV-0734 and one in series with CV-0735 to meet single failure criteria which will also close on steam generator low pressure. This modification, along with the existing system means that on a steam generator low pressure signal following a main steam line break, the following valves will trip closed: the feedwater reg valves, the existing bypass valves, the main feedwater stop valve and the new bypass valves."

In response to a NRC request for additional information regarding the potential consequences of such an event and the compensatory measures that have been taken to cope with the event until the proposed modifications can be completed, the Licensee stated [9]:

"Consumers Power Company letters dated October 26, 1982 [7] and July 22, 1983 [8] evaluated the consequences of such an event and found that the potential exists for exceeding the containment design pressure (55 psig). Although no analysis has been performed to precisely quantify the peak containment pressure that would result, it is apparent for the reasons listed below that containment pressure would not exceed that predicted for the blowdown of both steam generators, which the NRC staff has concluded is acceptable based on containment design margins for the specific issue of two steam generator blowdown due to main steam line isolation valve failure (ref. NUREG-0820 - Integrated Plant Safety Assessment - Palisades Plant, October, 1982 [11]). These reasons are:

1. The continued feedwater addition concern is only applicable if offsite power is available. In that case substantially more containment heat removal equipment (two spray pumps and four air coolers as compared to one spray pump and three air coolers) would be available to limit the peak containment pressure; and

2. The rate of mass and energy release to containment will be substantially less than that for a two steam generator blowdown.

Our letter of April 26, 1982 [6] described the capabilities of the main feedwater isolation system as presently designed. The system consists of isolation valves in each main feed line and bypass line which close automatically on low steam generator pressure. The possible single failure of a valve to close is addressed in the emergency operating procedures wherein the operator is instructed, as part of his immediate actions, to check the position of the main feedwater regulating and regulating bypass valves to verify that they are in fact closed. If a valve did not close, the operator would attempt to isolate the main feedwater system by manual means including closing the main feedwater regulating and block valves and the main feedwater bypass regulating valve either from the control room or at the valve itself."

3.1.2 Evaluation

The Licensee's submittals concerning the containment pressure response following a MSLB [3, 6-9], applicable sections of the Palisades FSAR [10], and the CPC letter [12] were reviewed in order to evaluate whether the following portions of the acceptance criteria were met:

- o Criterion 1.a - Continuation of flow to the affected steam generator
- o Criterion 1.b - Potential for containment overpressure
- o Criterion 1.c - Ability to detect and isolate the damaged steam generator
- o Criterion 4 - Potential for AFW pump damage
- o Criterion 5 - Design of steam and feedwater isolation system
- o Criterion 6 - Decay heat removal capacity
- o Criterion 7 - Safety-grade requirements for MFW and AFW isolation valves.

In reviewing the MSLB analysis, the Licensee discovered a design oversight that caused excessive condensate flow to the ruptured steam generator. In order to ensure complete isolation of the MFW and condensate flows, the Licensee modified the MFW regulating valves to close on low steam generator pressure. In addition, the Licensee [8] has made a commitment to modify the MFW stop and bypass valves to close on low steam generator pressure.

The Licensee's previous analysis did not account for the impact of runout AFW flow. Since the AFW system was manually initiated, it was assumed that the operator would feed only the unaffected steam generator. The Licensee then modified the plant's AFW system to provide for a control-grade automatic actuation system.

If a MSLB were to occur, the following sequence of events would follow:

- o Low steam generator pressure signal would cause:
    - the main steam isolation valves to shut
    - a reactor trip
    - the MFW stop, regulating, and bypass valves to shut.
- o Safety injection signal would be produced.
- o Auxiliary feedwater would be initiated after a 2-minute delay.

The initiating signals and circuits of the above systems were designed to meet IEEE Std 279-1971.

The Licensee performed a new analysis to evaluate the effect of auxiliary feedwater on the containment response to a MSLB. This analysis assumed a double-ended MSLB at the steam generator nozzle at 102% of full power and complete MFW isolation at 60 seconds. The most restrictive single failure was identified by the Licensee as a loss of offsite power with the failure of a diesel generator, which resulted in the loss of two out of three containment spray pumps and one out of four containment coolers. In this situation, at the 60-second point, full containment spray flow from one pump is developed. The energy removal rate of the containment spray and air coolers then exceeds the energy addition rate due to the MSLB. The containment pressure peaks at 68.5 psia (53.8 psig), which is below the design pressure of 55 psig, and the containment starts to depressurize. AFW initiation is delayed until 120 seconds after the MSLB, at which time the AFW flow control valve is assumed to fail, allowing a runout flow of 750 gpm to the affected steam generator. Initiation of runout AFW flow will not affect the peak containment pressure but will reduce the rate of containment depressurization.

In a letter to the NRC [12], the Licensee described further modifications to the AFW system. These modifications served to meet the requirements of NUREG-0737 regarding automatic initiation of the AFW system. The modified AFW system is designed to detect and redundantly isolate a ruptured steam generator and to provide adequate AFW flow to the unaffected steam generator to ensure that system heat removal capacity exceeds the minimum level required for decay heat removal. The NRC's review of the AFW system [22] determined that the system complied with the long-term safety-grade requirements of NUREG-0737.

With the AFW system proposed design to isolate the affected steam generator from all AFW flow, the Licensee's current MSLB analysis will bound the present plant design. With these modifications, the AFW pumps will not experience runout flow and will therefore perform their function during a MSLB accident without incurring damage.

The environmental qualification of safety-related electrical and mechanical components is being reviewed separately by the NRC and is not within the scope of this review.

The review did not determine if the instrumentation that the operator relies upon to follow the accident and isolate the affected steam generator conforms with the criteria in ANS/ANSI 4.5-1980 and Regulatory Guide 1.97.

In a separate analysis performed as part of the Integrated Plant Safety Assessment and reported in NUREG-0820 [11], the most restrictive single failure was that of a main steam isolation valve in conjunction with a MSLB inside containment, which would result in blowdown of both steam generators, producing a peak pressure of 107 psia or 153% of the design pressure of 55 psig. A further analysis was performed assuming a fix to prevent the blowdown of both steam generators, which produced results similar to the Licensee analysis [3]. The NRC's review [11] of this overpressure condition determined that there was sufficient margin in the containment design so that no damage would occur to the containment structure. The issue of modifying the main steam system to ensure that only one steam generator will blow down in the event of a MSIV failure is being handled separately by the NRC and is not within the scope of this review.

3.1.3 Conclusion and Recommendation

The Licensee's responses [3, 6-9], the Palisades FSAR [10], and Reference 12 adequately address the concerns of Item 1 of IE Bulletin 80-04. The containment pressure response analysis and the design of the mitigating systems satisfy the NRC's acceptance criteria. It is concluded that there is no potential for containment overpressurization resulting from a MSLB with continued feedwater addition. In addition, since the AFW pumps do not experience runout conditions, the pumps will be able to carry out their intended function without incurring damage.

3.2 REVIEW OF REACTIVITY INCREASE ANALYSIS

The requirement from IE Bulletin 80-04, Item 2, is as follows:

"Review your analysis of the reactivity increase which results from a main steam line break inside or outside containment. This review should consider the reactor cooldown rate and the potential for the reactor to return to power with the most reactive control rod in the fully withdrawn position. If your previous analysis did not consider all potential water sources (such as those listed in 1 above) and if the reactivity increase is greater than previous analysis indicated the report of this review should include:

a. The boundary conditions for the analysis, e.g., the end of life shutdown margin, the moderator temperature coefficient, power level and the net effect of the associated steam generator water inventory on the reactor system cooling, etc.,

b. The most restrictive single active failure in the safety injection system and the effect of that failure on delaying the delivery of high concentration boric acid solution to the reactor coolant system,

c. The effect of extended water supply to the affected steam generator on the core criticality and return to power,

d. The hot channel factors corresponding to the most reactive rod in the fully withdrawn position at the end of life, and the Minimum Departure from Nucleate Boiling Ratio (MDNBR) values for the analyzed transient."

3.2.1 Summary of Licensee Statements and Conclusions

In regard to the reactivity increase resulting from a MSLB with continued feedwater addition, the Licensee stated [3]:

"Consumers Power Company has reviewed the analysis of the reactivity increase which would result from an MSLB inside or outside containment. It has been determined that, although the impact of runout flow from the auxiliary feedwater system was not considered in the analysis, the analysis adequately bounds this case. The applicable MSLB analyses are located in the following references:

Full Power, Off-Site Power, Inside Containment - Ref 1, Section 3.8.1
Full Power, No Off-Site Power, Inside Containment - FSAR, Amend 17, Section 5.0
Full Power, No Off-Site Power, Outside Containment - FSAR, Amend 15, Section 14.3
Zero Power, Off-Site Power, Inside Containment - Ref 1, Section 3.8.2

Reference 1 [23]: XN-NF-77-18, 'Plant Transient Analysis of the Palisades Reactor for Operation at 2530 MWt.'

In each full power case, the main feedwater flow to each steam generator was assumed to be reduced from full flow to 5% of full flow (the mass equivalent of 560 gpm of cold auxiliary feedwater to each steam generator) over the 60 seconds immediately following a reactor trip. This assumption is conservative when considering runout flow from one auxiliary feedwater pump for the following reasons:

a. The plant is presently being modified as a result of a discovered deficiency (see Licensee Event Report 79-041) to close the main feedwater regulating and bypass valves on a low steam generator pressure signal (approximately 500 psia). This will result in a complete termination of main feedwater much sooner than assumed in the analysis. The analysis also assumed more than 48,000 lbm of main feedwater to be delivered to the broken steam generator during the first 60 seconds.

b. In each case, the peak core heat flux and the MDNBR occurs at or before the time of steam generator dryout (where dryout is defined to occur when break flow = assumed feedwater flow), and before the initiation of auxiliary feedwater at 120 seconds. Steam generator dryout causes an abrupt drop in steam flow, core power and an increase in MDNBR. A slightly greater feedwater flow rate and steam flow rate after steam generator dryout (i.e., 750 gpm instead of 560 gpm) would have no significant impact on criticality margins or core power levels.

For the zero power case, 415 gpm of cold auxiliary feedwater were assumed to be delivered to each steam generator for the duration of the analyzed transient.

This assumption is bounding even when considering runout from one auxiliary feedwater pump for the following reason. The safety analysis shows that the peak core heat flux occurs long before the time of steam generator dryout. A slightly higher feedwater flow rate (750 gpm vs 415 gpm) would not significantly affect the magnitude or the rate of primary system cooldown (or the resulting criticality margin) prior to dryout. The cooldown rate is chiefly governed by the break area, the primary-to-secondary temperature differential, and the primary coolant flow rate.

This small amount of additional auxiliary feedwater prior to dryout would have a negligible effect on secondary temperature and an insignificant effect on the core return to power."

In response to a request regarding the effect of continued MFW flow on the reactivity response, the Licensee stated [7]:

"The reactor response to a MSLB was analyzed by Exxon and reported in XN-NF-77-18. The most severe MSLB was determined to be from 102% of rated power with a resulting Minimum Critical Heat Flux Ratio (MCHFR) of 1.30. MCHFR occurs at the time of maximum core average heat flux. From the above report, after about 40 seconds following the MSLB the core average heat flux increases and continues to increase with the concentrated boric acid from the charging pumps reaches the core at 96 seconds. From that time until the steam generator empties at 126 seconds the core average heat flux remains constant after which it decreases. With continued feedwater, the time until the steam generator empties will be lengthened. However, the peak core average heat flux will be unchanged since it is not a function of the time that the steam generator empties. Thus, continued feedwater will not result in a lower MCHFR."

3.2.2 Evaluation

The Licensee's analysis of the core reactivity increase resulting from a MSLB with continued feedwater addition was reviewed to determine whether the following acceptance criteria were met:

- o Criterion 1.c - Ability to detect and isolate the damaged steam generator
- o Criterion 1.d - Changes in core reactivity increase
- o Criterion 3 - Analysis assumptions.

From review of the FSAR [10] and Exxon analyses [23] of the reactivity increase resulting from a MSLB, it was determined that the assumptions of the analyses were conservative and in accordance with those in Acceptance Criterion 3.

The Exxon analysis [23] determined that, for the worst-case scenario (zero power, double-ended MSLB at the steam generator nozzle, 415 gpm AFW flow, loss of one safeguards train), neither a return-to-power nor a violation of the specified acceptable fuel design limits occurs.

In view of the MFW and AFW system modifications to provide isolati n of MFW and AFW flow to a ruptured steam generator, the assumptions regarding MFW and AFW flow are conservative. Therefore, the Licensee's current MSLB analysis [23] bounds those assumptions required for the current plant design.

3.2.3 Conclusion and Recommendation

The Licensee's response, the FSAR, and subsequent analysis adequately address the concerns of Item 2 of IE Bulletin 80-04. All potential sources of water were identified and are isolated. No return-to-power is predicted, and no there is violation of the specified acceptable fuel design limits; therefore, the current analysis [23] remains valid and no further action is required.

3.3 REVIEW OF CORRECTIVE ACTIONS

The requirement from IE Bulletin 80-04, Item 3, is as follows:

> "If the potential for containment overpressure exists or the reactor return-to-power response worsens, provide a proposed corrective action and a schedule for completion of the corrective action. If the unit is operating, provide a description of any interim action that will be taken until the proposed corrective action is completed."

3.3.1 Summary of Licensee Statements and Conclusions

The Licensee made a commitment to make the following modifications [8]:

- o Modify the circuitry of the MFW stop valves and MFW bypass valves to close on low steam generator pressure trip signal.
- o Install additional control valves in series with the MFW stop valves and bypass valves which will also close on low steam generator pressure signal.

These modifications will be completed during the 1985 refueling outage.

3.3.2 Evaluation, Conclusion, and Recommendation

The addition of the valves in series with the MFW stop valves and MFW bypass valves, all closing on receipt of a low steam generator pressure signal, provides single-failure-proof protection against the continued addition of main feedwater during a MSLB. The Licensee's analysis determined that neither a containment overpressurization nor a reactor return-to-power and violation of the specified acceptable fuel design limits would result from a MSLB. Therefore, it was concluded that no further action regarding IE Bulletin 80-04 is required of CPC for the Palisades Plant.

4. CONCLUSIONS

Conclusions regarding Consumers Power Company's response to IE Bulletin 80-04 with respect to Palisades Plant are as follows:

- There is no potential for containment overpressurization resulting from a MSLB with continued feedwater addition.
- The AFW pumps will not experience runout conditions and will therefore be able to carry out their intended function without incurring damage during a MSLB.
- All potential water sources were identified, no return-to-power is predicted, and there is no violation of the specified acceptable fuel design limits. Therefore, the current analysis remains valid.
- No further action by the Licensee is required regarding IE Bulletin 80-04.

5. REFERENCES

1. "Analysis of a PWR Main Steam Line Break with Continued Feedwater Addition"
   NRC Office of Inspection and Enforcement, February 8, 1980
   IE Bulletin 80-04

2. "Overpressurization of the Containment of a PWR Plant after a Main Line Steam Break"
   NRC, Office of Inspection and Enforcement, October 1, 1979
   IE Information Notice 79-24

3. S. R. Frost (CPC)
   Letter to J. G. Keppler (NRC, Region III)
   Subject: Response to IE Bulletin 80-04
   9-May-80

4. D. M. Crutchfield (NRC)
   Letter to Consumers Power Co.
   Subject: IE Bulletin 80-04, Request for Additional Information
   March 8, 1982

5. D. M. Crutchfield (NRC)
   Letter to Consumers Power Co.
   Subject: IE Bulletin 80-04, Request for Additional Information
   September 7, 1982

6. B. D. Johnson (CPC)
   Letter to D. M. Crutchfield (NRC)
   Subject: Response to IE Bulletin 80-04 Additional Information
   April 26, 1982

7. B. D. Johnson (CPC)
   Letter to D. M. Crutchfield (NRC)
   Subject: Analysis of Main Steam Line Break with Continued Feedwater Addition
   October 26, 1982

8. B. D. Johnson (CPC)
   Letter to J. G. Keppler (NRC IE Region III)
   Subject: Analysis of Main Steam Line Break with Continued Feedwater Addition
   July 22, 1983

9. B. D. Johnson (CPC)
   Letter to D. M. Crutchfield (NRC)
   Subject: Analysis of Main Steam Line Break with Continued Feedwater Addition
   September 8, 1983

10. Palisades Plant Final Safety Analysis Report
Consumers Power Company

11. Integrated Plant Safety Assessment
Palisades Plant
NUREG-0820
October 1982

12. B. D. Johnson (CPC)
Letter to D. M. Crutchfield (NRR)
Subject: NUREG-0737, Item II.E.1.1 - Additional Information
May 29, 1981

13. Technical Evaluation Report, "PWR Main Steam Line Break with Continued Feedwater Addition - Review of Acceptance Criteria"
Franklin Research Center, November 17, 1981
TER-C5506-119

14. "Criteria for Protection Systems for Nuclear Power Generating Stations"
Institute of Electrical and Electronics Engineers, New York, NY, 1971
IEEE Std 279-1971

15. Standard Review Plan, Section 4.2
"Fuel System Design"
NRC, July 1981
NUREG-0800

16. Standard Review Plan, Section 15.1.5
"Steam System Piping Failures Inside and Outside of Containment (PWR)"
NRC, July 1981
NUREG-0800

17. "Criteria for Accident Monitoring Functions in Light-Water-Cooled Reactors"
American Nuclear Society, Hinsdale, IL, December 1980
ANS/ANSI-4.5-1980

18. "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident," Rev. 2
NRC, December 1980
Regulatory Guide 1.97

19. "Single Failure Criteria for PWR Fluid Systems"
American Nuclear Society, Hinsdale, IL, June 1976
ANS-51.7/N658-1976

20. "Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants"
Rev. 3
NRC, February 1976
Regulatory Guide 1.26

21. "Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment," Rev. 1
NRC, July 1981
NUREG-0588

22. Technical Evaluation Report, "Auxiliary Feedwater System Automatic Initiation and Flow Indication"
Franklin Research Center, April 1982
TER-5257-298

23. "Plant Transient Analysis of the Palisades Reactor for Operation at 2530 MWt"
Exxon Nuclear Corporation
XN-NF-77-18