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TECHNICAL EVALUATION OF THE ELECTRICAL, INSTRUMENTATION, AND CONTROL DESIGN ASPECTS OF THE PROPOSED LICENSE AMENDMENT FOR SINGLE-LOOP OPERATION OF THE MONTICELLO NUCLEAR GENERATING PLANT, UNIT 1

(DOCKET NO. 50-263)

APRIL 1980

Work Performed for Lawrence Livermore Laboratory under U.S. Department of Energy Contract No. DE-AC08-76 NVO 1183.

EG&G
Energy Measurements Group
San Ramon Operations

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by

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ABSTRACT

This report documents the technical evaluation of the electrical, instrumentation, and control design aspects of the proposed license amendment for single-loop operation of the Monticello nuclear generating plant, Unit 1. The review criteria are based on IEEE Std-279-1971 requirements and the Code of Federal Regulations, Title 10, Part 50.46 and Part 50, Appendices A and K requirements for single-loop operation. This report is supplied as part of the Selected Electrical, Instrumentation, and Control Systems Issues Program being conducted for the U. S. Nuclear Regulatory Commission by Lawrence Livermore Laboratory.

FOREWORD

This report is supplied as part of the Selected Electrical, Instrumentation, and Control Systems Issues (SEICSI) Program being conducted for the U. S. Nuclear Regulatory Commission, Office of Nuclear Reactor Regulation, Division of Operating Reactors, by Lawrence Livermore Laboratory, Field Test Systems Division of the Electronics Engineering Department.

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(Docket No. 50-263)

James H. Cooper
EG&G, Inc., Energy Measurements Group, San Ramon Operations

1. INTRODUCTION

By letter[1] to the U. S. Nuclear Regulatory Commission (NRC) dated September 7, 1976, the Northern States Power Company (NSPC) submitted information to support its proposed license amendment to operate the Monticello nuclear generating plant, Unit 1, with one recirculation loop out of service (i. e., single-loop operation). This information represented the licensee's analysis of significant events, based on a review of accidents and abnormal operational transients associated with power operations in the single-loop mode and provided by the nuclear steam supply system designer (General Electric Company, Nuclear Energy Division (GE-NED)). Conservative assumptions were employed in the GE-NED Report NEDO-21252,[2] dated June 1976, to ensure that the generic analyses for boiling water reactors (BWR 3 and/or 4) were applicable to the Monticello nuclear generating plant, Unit 1. GE-NED submitted an additional report (NEDO-20566-2,[3] dated July 1978) of an analytical model for a loss-of-coolant accident (LOCA) with one recirculation loop out of service which is under review by the NRC Reactor Safety Branch (RSB).

The purpose of this report is to evaluate the electrical, instrumentation, and control (EI&C) design aspects of the proposed license amendment as presented in NEDO-21252[2] and using IEEE Std-279-1971[4] criteria and the Code of Federal Regulations, Title 10, Part 50.46,[5] and Title 10, Part 50, Appendix A[6] and Appendix K[7] criteria.

2. DESCRIPTION AND EVALUATION OF THE PROPOSED LICENSE AMENDMENT FOR SINGLE-LOOP OPERATION

2.1 DESCRIPTION OF THE PROPOSED CHANGES

The licensee states that, from its analysis of NEDO-21252,[2] the only changes necessary to the reactor protection system (RPS) are modifications to the:

(1) SCRAM trip settings of the average power range monitor (APRM) system.

(2) Rod-block setpoints of the rod block monitor (RBM) system.

Because of the different flow quantity and different flow path during single-loop operation, the APRM SCRAM trip settings, which are flow-biased according to the equation in the technical specifications, require re-setting to protect the reactor from overpower. The rod-block setpoint equation is flow-biased in the same way and with the same flow signal as the APRM setpoint, and must also be modified to provide adequate local core protection for the postulated rod withdrawal error.

The revised technical specifications propose single-loop operation at reduced safety settings for unlimited periods of time. The revised technical specifications also propose a limit of 24 hours in which to reduce the safety settings. Use of Section 3.4.1.1.a of the standard BWR technical specifications[8] will be required. Section 3.4.1.1.a states that

> "With one recirculation loop not in operation, (reactor) operation may continue; restore both loops to operation within 12 hours or be in at least hot shutdown within the next 12 hours."[8]

The numerical values of the new settings are delineated in the revised technical specifications which accompany the licensee's submittal.[1]

2.2 EVALUATION OF THE PROPOSED CHANGES

The temporary changes in the settings of the trip points for the APRM and RBM must be made in the power-range cabinets in the control room and so must be done with the reactor shut down (i. e., with the mode switch in shutdown or refuel, condition 3, 4, 5) as required by the NRC Branch Technical Position ICSB 12.[9] These adjustments include readjusting the power and flow potentiometers in each of the six APRM channels and the two RBM channels. One channel of the multichannel systems will be adjusted at a time and then returned to service. Before all of the channels are returned to service, the new trip setpoints will be verified by the instrument engineer following the readjustment and testing of the setpoints by the instrument technician. Two operators will perform functional tests to double check the new setpoints and to check the instrument's return to an operable condition.

The sequence outlined above shall be written into the plant technical specifications. A permanent installation of the setpoint-change capability must be made in order for the system to satisfy the requirements of Section 4.15 of IEEE Std-279-1971.[4] Hard-wire modifications will be required to enable making setpoint changes from the front panel of the power range cabinet by way of control switches. The licensee's proposed modifications must be submitted to the NRC staff for review prior to this installation.

The recirculation-loop equalizer valves must be verified closed and tagged for single-loop power operation, as is the case for two-loop power operation. The safety analysis in NEDO-21252[2] assumes that these valves remain closed as their effect on a LOCA has not been analyzed.

Recirculation flow must be manually controlled by the operator, as opposed to automatic control, whenever the system is operating in the single-loop mode, since control stability is degraded and manual control is assumed in the NEDO-21252.[2] The technical specifications will be changed to include this restriction.

Due to the different flow pattern during single-loop operation as described by the licensee, a number of indications in the control room will change, such as individual jet-pump flow and total summed core flow. Some indications will be only slightly less accurate, but some others will be erroneous. The control room indications must be corrected prior to single-loop power operation or they must be tagged out-of-service, as appropriate. This is a requirement of Section 4.20 of IEEE Std-279-1971.[4]

The normal plant configuration as described in the final safety analysis report (FSAR)[10] includes recirculation-pump start interlocks to prevent an inadvertent cold-water injection into the reactor. Any recirculation loop that is out of service and whose water has cooled must be run in the bypass mode to preheat the water to within 100°F of the reactor cooling water before the water may be valved back to the reactor pressure vessel. The recirculation pump start is interlocked to permit start-up only if the pump discharge valve is closed, the bypass valve is open, and the suction valve is open. This configuration will limit the amount of cold water which can be transported through the reactor vessel from a cold-loop startup, thereby limiting the effect of a cold-water slug event. Although interlocks are provided, no credit is taken for their safety function in NEDO-21252[2] for single-loop operation since this is not the limiting transient.

The instrument setpoints can be set down to enable operation in the single-loop mode for unrestricted periods. This mode of operation is desired by the licensee to facilitate more extensive unscheduled maintenance without the requirement of keeping the reactor shut down. It is stipulated that single-loop operation will not be a planned mode of operation.

The new rod-block and trip setpoints vary linearly as a function of recirculation flow rate. For power increases by rod withdrawal, the RBM rod block must be set to the next higher trip level by manual operator action. The APRM, flow-biased, SCRAM trip follows the new trip curve automatically for both power increases and decreases.

3. CONCLUSIONS

3.1 INTRODUCTION

The initial licensee submittal analyzed in this report was based on the analysis in NEDO-21252[2] performed by the nuclear steam supply system manufacturer (GE-NED). The manufacturer, however, had not analyzed the performance of the emergency core cooling system (ECCS) during single-loop operation.

A new analysis was performed by GE-NED for a LOCA with one recirculation loop out of service. The analysis, reported in NEDO 20566-2,[3] includes the ECCS single loop analysis. This NEDO report was published subsequent to the initial licensee submittal. The licensee will submit more detailed information based on the new analysis.

3.2 CONCLUSIONS

We conclude that the concept outlined in the Northern States Power company's initial license amendment submittal[1] is in accordance with the intent of IEEE-Std-271-1971. At the conclusion of this review, the specific design criteria noted herein were communicated to the licensee, as required by the NRC. The licensee has agreed to address that criteria in the detailed license amendment submittal which was necessitated by the revised GE report. We recommend that the NRC review the EI&C aspects of the forthcoming detailed license amendment submittal for conformance to these criteria.

REFERENCES

1. NSPC Letter to NRC (V. Stello), dated September 7, 1976.

2. General Electric Company, Nuclear Energy Division, License Amendment Submittal for Single-Loop Operation - Monticello Nuclear Generating Plant, Unit 1, NEDO-21252 (June 1976).

3. General Electric Company Nuclear Energy Division, An Analytical Model for Loss-of-Coolant Accident (LOCA) with One Recirculation Loop Out-Of-Service, NEDO-20566-2 (July 1978).

4. IEEE Std-279-1971: Criteria for Protection Systems for Nuclear Power Generating Stations (n. d.).

5. Code of Federal Regulations, Title 10, Part 50.46: Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Reactors (January 1976).

6. Code of Federal Regulations, Title 10, Part 50, Appendix A: General Design Criteria for Nuclear Power Plants (January 1978).

7. Code of Federal Regulations, Title 10, Part 50, Appendix K: ECCS Evaluation Models (January 1, 1978).

8. General Electric Company, Standard Boiling Water Reactor Technical Specifications (April 1978).

9. NRC/RSB, Protection System Trip Point changes for Operation With Reactor Coolant Pump Out of Service, Branch Technical Position ICSB 12 (n. d.).

10. NSPC, Final Safety Analysis Report for Monticello Nuclear Power Station (FSAR) (n. d.).

ENCLOSURE

SAFETY EVALUATION REPORT N-1 LOOP OPERATION

MONTICELLO NUCLEAR GENERATING PLANT

1.0 INTRODUCTION

The current Monticello Technical Specifications do not allow plant operation beyond 24 hours if an idle recirculation loop can not be returned to service. The ability to operate at reduced power with a single loop is highly desirable from an availability/outage planning standpoint in the event that maintenance or component unavailability renders one loop inoperable.

By letter dated July 2, 1982, Northern States Power Company (NSP) (the licensee) requested changes to the Technical Specification for Single Loop Operation of Monticello. The requested changes would permit Monticello to operate at up to 50% of rated power with one recirculation loop out of service for unlimited time. While analyses indicate that it may be safe to operate BWRs on a single loop in the range higher than 50% of rated power, the experience (reference letter from L. M. Mills, TVA dated March 17, 1980 to H. Denton, NRC) at Browns Ferry Unit 1 has caused concern about flow and power oscillations. However, because single loop operation at 50% rated power at several plants, including Browns Ferry Plant Unit-1, has shown to result in acceptable flow and power characteristics, we will permit NSP to operate at power levels up to 50% of rated with one loop out of service during Cycle 10.

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If requested, we will reconsider operation at a higher power level for Monticello with one recirculation loop out of service after staff concerns stemming from Browns Ferry - Unit 1 single loop operation are satisfied.

2 EVALUATION

2.1 Accidents (Other than Loss of Coolant Accident (LOCA)) and Transients Affected by One Recirculation Loop Out of Service

2.1.1 One Pump Seizure Accident

The licensee states that the one-pump seizure accident is a relatively mild event during two recirculation pump operation. Similar analyses were performed to determine the impact this accident would have on one recirculation pump operation. These analyses were performed using NRC approved models for a large core BWR/4 plant. The analyses were conducted from steady-state operation at the following initial conditions, with the added condition of one inactive recirculation loop. Two sets of initial conditions were assumed:

a. Thermal Power = 75% and core flow = 58% of rated

b. Thermal Power = 82% and core flow = 56% of rated

These conditions were chosen because they represent reasonable upper limits of single-loop operation within existing Maximum Average Linear Heat Generation Rate (MAPLHGR) and Minimum Critical Power Ratio (MCPR) limits at the same maximum pump speed. Pump seizure was simulated by setting the single operating pump speed to zero instantaneously.

The anticipated sequence of events following a recirculation pump seizure which occurs during plant operation with the alternate recirculation loop out of service is as follows:

a. The recirculation loop flow in the loop in which the pump seizure occurs drops instantaneously to zero.

b. Core voids increase which results in a negative reactivity insertion and sharp decrease in neutron flux.

c. Heat flux drops more slowly because of the fuel time constant.

d. Neutron flux, heat flux, reactor water level, steam flow, and feedwater flow all exhibit transient behaviors. However, it is not anticipated that the increase in water level will cause a turbine trip and result in scram.

It is expected that the transient will terminate at a condition of natural circulation and an orderly reactor shutdown will be accomplished. There will also be a small decrease in system pressure.

The licensee concludes that the MCPR for the pump seizure accident for the large core BWR/4 plant was determined to be greater than the fuel cladding integrity safety limit; therefore, no fuel failures were postulated to occur as a result of this analyzed event. These results are applicable to Monticello.

2.1.2 Abnormal Transients

2.1.2.1 a. Idle Loop Startup

The idle loop startup transient was anlayzed, in the Monticello FSAR, with an initial power of 60%. The licensee is to operate at no greater than 50% power with one loop out of service. Additionally, the Technical Specifications are being modified to require that, during single loop operation, the suction valve in the idle loop be shut and electrically disconnected. These measures are being taken to preclude startup of an idle loop.

b. Flow Increase

For single-loop operation, the rated condition steady-state MCPR limit is increased by 0.01 to account for increased uncertainties in the core total flow and Traversing In-core Probe (TIP) readings. The MCPR will vary depending on flow conditions. This leads to the possibility of a large inadvertent flow increase which could cause the MCPR to decrease below the Safety Limit for a low initial MCPR at reduced flow conditions. Therefore, the required MCPR must be increased at reduced core flow by a flow factor K_f. The K_f factors are derived assuming both recirculation loop pumps increase speed to the maximum permitted by the scoop tube position set screws. This condition maximizes the power increase and hence maximum Δ MCPR for transients initiated from less than rated conditions. When operating on one loop the flow and power increase will be less than associated with two pumps increasing speed, therefore, the K_f factors derived from the two-pump assumption are conservative for single-loop operation.

c. Rod Withdrawal Error

The rod withdrawal error at rated power is given in the FSAR for the inital core and in cycle dependent reload supplemental submittals. These analyses are performed to demonstrate that, even if the operator ignores all instrument indications and the alarm which could occur during the course of the transients, the rod block system will stop rod withdrawal at a minimum critical power ratio which is higher than the fuel cladding integrity safety limit. Correction of the rod block equation and lower initial power for single-loop operation assures that the MCPR safety limit is not violated.

One-pump operation results in backflow through 10 of the 20 jet pumps while flow is being supplied to the lower plenum from the active jet pumps. Because of this backflow through the inactive jet pumps the present rod-block equation and APRM settings must be modified. The licensee has modified the two-pump rod block equation and APRM settings that exist in the Technical Specification for one-pump operation and the staff has found them acceptable.

The staff finds that one loop transients and accidents other than LOCA, which is discussed below, are bounded by the two loop operation analysis and are therefore acceptable.

2.2 Loss of Coolant Accident (LOCA)

The licensee has contracted General Electric Co. (GE) to perform single loop operation analysis for Monticello LOCA. The licensee states that evaluation of these calculations (that are performed according to the procedure outlined in NEDO-20566-2, Rev. 1) indicates that a muliplier of 0.85 (8x8 fuel), 0.85 (8x8R Fuel) 0.85 (P8x8R fuel) (Ref: - NEDE 24271 June 1980) should be applied to the MAPLHGR limits for single loop operation of Monticello.

We find the use of these MAPLHGR multipliers to be acceptable.

3. THERMAL HYDRAULICS

The licensee has confirmed that analysis uncertainties are independent of whether flow is provided by two loops or single loop. The only exceptions to this are core total flow and TIP reading. The effect of these uncertainties is an increase in the MCPR by .01, which is more than offset by the K_f factor required at low flows. The steady state operating MCPR with single-loop operation will be conservatively established by multiplying the K_f factor to the rated flow MCPR limit.

4. STABILITY ANALYSIS

As indicated in the applicant's submittal NEDO-24271, operating along the minimum forced recirculation line with one pump running at minimum speed is more stable than operating with natural circulation flow only, but is less stable than operating with both pumps operating at minimum speed.

The licensee will be required to operate in master manual to reduce the effects of instabilities due to controller feedback. The staff has accepted previous stability analyses results as evidence that the core can be operated safely while our generic evaluation of BWR stability characteristics and analysis methods continues. The previous stability analysis results include natural circulation conditions and thus bound the single loop operation. In addition, the decay ratio (.62) predicted for cycle 10 of Monticello shows margin relative to Browns Ferry #1 (.83) which had the flow noise oscillations during SLO. We conclude that with appropriate limitations to recognize and avoid operating instabilities, that the reactor can be operated safely in the single loop mode. Our evaluation of the flow/power oscillations evidenced in Browns Ferry will continue and any pertinent conclusions resulting from this study will be applied to Monticello.

5. SUMMARY ON SINGLE LOOP OPERATION

1. Steady State Thermal Power Level will not exceed 50%

Operating at 50% power with appropriate T-S changes was approved on a long term basis for the Duane Arnold Plant, Peach Bottom Units 2 and 3, Pilgrim-1 and Cooper Nuclear Station (Safety Evaluation Reports (SER) dated November 19, 1981, May 15, 1981, December 15, 1981, and December 10, 1981 respectively). Authorization for single loop operation for extended periods was also given to Dresden Unit 2 and 3 and Quad Cities Units 1 and 2 (SER July 9, 1981). It was concluded that for operation at 50%, power transient and accident bounds would not be exceeded for these plants.

2. Minimum Critical Power Ratio (MCPR) Safety Limit will be Increased by 0.01 to 1.08

The MCPR Safety Limit will be increased by 0.01 to account for increased uncertainties in core flow and Traversing Incore Probe (TIP) readings. The licensee has reported that this increase in the MCPR Safety Limit was addressed in GE reports specifically for Monticello for one loop operation. On the basis of previous staff reviews for Pilgrim-1, Cooper, Duane Arnold and Peach Bottom and our review of plant comparisons we find this analysis acceptable for Monticello.

3. Minimum Critical Power Ratio (MCPR) Limiting Condition for Operation (LCO) will be Increased by 0.01.

The staff require that the operating limit MCPR be increased by 0.01 and multiplied by the appropriate two loop K_f factors that are in the Monticello T-S. This will preclude an inadvertent flow increase from causing the MCPR to drop below the safety limit MCPR. This was also approved by the staff for Peach Bottom 2 and 3.

4. The Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) Limits will be Reduced by Appropriate Multipliers

The licensee proposed reducing the T-S MAPLHGR by 0.85 (8x8 fuel); 0.85 (8x8R) and 0.85 (P8x8R) for single loop operation. These reductions were based on analyses by General Electric (GE) in reports NEDE 24011-P-A-1 and NEDO 24271. The Peach Bottom units were allowed to operate with their MAPLHGR values reduced by factors of 0.71, 0.83, and 0.81 for an unlimited period of time for the first three types of fuel listed above.

5. The APRM Scram and Rod Block Setpoints will be Reduced

The licensee proposed to modify the two loop APRM Scram, Rod Block and Rod Block Monitor (RBM) setpoints to account for back flow through half the jet pumps. The changes were based on plant specific anlayses by GE. These setpoints equations will be changed in the Monticello T-S. The above changes are similar to the Peach Bottom T-S changes and are acceptable to the staff.

6. The Suction Valve in the Idle Loop is Closed and Electrically Isolated

The licensee will close the recirculation pump suction valve and remove power from the valve. In the event of a loss of coolant accident this would preclude partial loss of LPCI flow through the recirculation loop degrading the intended LPCI performance. The removal of power also helps to preclude a start up of an idle loop transient.

7. The Equalizer line between the loops will be Isolated

The licensee will close appropriate valves in the cross-tie (equalizer) line between the loops. The previously discussed analysis assumed the two loops were isolated. Therefore, it is required that the cross-tie valve be closed.

8. The Recirculation Control will be in Manual Control

The staff requires that the licensee operate the recirculation system in the manual mode to eliminate the need for control system analyses and to reduce the effects of potential flow instabilities. This was also required of Peach Bottom.

9. Surveillance Requirements

The staff requires that the licensee perform daily surveillance on the jet pumps to ensure that the pressure drop for one jet pump in a loop does not vary from the mean of all jet pumps in that loop by more than 5%.

10. Provisions to Allow Operation with One Recirculation Loop out of Service

1. The steady-state thermal power level will not exceed 50% of rated

2. The Minimum Critical Power Ratio (MCPR) Safety Limit will be increased by .01 to 1.08 (T.S. 3.11C)

3. The MCPR Limiting Condition for Operation (LCO) will be increased by 0.01

4. The Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) limits will be reduced by multiplying 0.85 for 8x8, 8x8R, P8x8R fuel respectively (T.S. reference Table 3.11.1)

5. The APRM Scram and Rod Block Setpoints and the RBM Setpoints, shall be reduced to read as follows:

T.S. 2.3.A.1 $S \leq (.65W + 55\% -0.65 \Delta W)$

T.S. 2.3.A.1* $S \leq (.65W + 55\% -0.65 \Delta W)$ FRP/MFLPD

T.S. 2.3.B $S \leq (.65W + 43\% -.65 \Delta W)$

T.S. 2.3.B* $S \leq (.65W + 43\% -0.65 \Delta W)$

APRM UPscale $\leq (.65 + 43\% -0.65 \Delta W)$

RBM Upscale $\leq (.65W +43\% - 0.65 \Delta W)$

*In the event that MFLPD exceeds FRP.

6. The suction valve in the idle loop is closed and electrically isolated until the idle loop is being prepared for return to service.

7. APRM flux noise will be measured once per shift and the recirculation pump speed will be reduced if the flux noise exceeds 5-percent peak to peak.

8. The core plate delta noise be measured once per shift and the recirculation pump speed will be reduced if the noise exceeds 1 psi peak to peak.

Therefore, based upon the above evaluation and a history of successful operation of other BWRs of the same type as Monticello we conclude that single-loop operation of Monticello up to a power level of 50% and in accordance with the proposed TSs, will not exceed the accident and transient bounds previously found acceptable by the NRC staff and is therefore acceptable.

The approval for single loop operation up to a power level of 50% is authorized during cycle 10 only.

We have concluded, based on the considerations discussed above, that: (1) because the amendment does not involve a significant increase in the probability or consequences of accidents previously considered or create the possibility of an accident of a type different from any evaluated previously, and does not involve a significant decrease in a safety margin, the amendment does not involve a significant hazards consideration, (2) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (3) such activities will be conducted in compliance with the Commission's regulations and the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.