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TECHNICAL EVALUATION OF THE ELECTRICAL, INSTRUMENTATION, AND CONTROL DESIGN ASPECTS OF THE PROPOSED LICENSE AMENDMENT FOR SINGLE-LOOP OPERATION OF COOPER NUCLEAR STATION

(Docket No. 50-298)

by

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SAN RAMON OPERATIONS

I. INTRODUCTION

By letter to the U.S. Nuclear Regulatory Commission (NRC) dated August 5, 1980 [Ref. 1], the Nebraska Public Power District submitted information to support its proposed license amendment to operate the Cooper Nuclear Station (CNS) with one recirculation loop out of service (i.e., single-loop operation). This information included the licensee's analysis of significant events, which were based on a review of accidents and abnormal operational transients associated with power operations in the single-loop mode provided by General Electric Company, Nuclear Energy Division (GE-NED), the nuclear steam supply system designer. Conservative assumptions were employed, as discussed in the GE-NED report NEDO-24258 dated May 1980 [Ref. 2], to ensure that the generic analyses for boiling water reactors (BWR 3 and/or 4) were applicable to the Cooper Nuclear Station.

In response to an NRC request, the licensee provided supplemental information in a letter dated May 6, 1982 [Ref. 3]. Subsequently, two telephone-conference calls were conducted with the licensee [Refs. 4 and 5] for further clarification of the protection system trip point setting changes for CNS single-loop operation. These were later documented by the licensee's letter dated July 28, 1982 [Ref. 6].

The purpose of this report is to document the evaluation of the electrical, instrumentation, and control (EI&C) design aspects of the proposed license amendment change to the CNS technical specifications. The consideration of proper plant variables, computer models, and the licensee's conclusions on core performance and clad temperature are outside the scope of this evaluation.

This review was conducted using 10CFR50, Appendix A, "General Design Criteria for Nuclear Power Plants" (G.D.C. 20 through 24) [Ref. 7] and ANSI/IEEE Std 279-1971 [Ref. 8] with the following guidance from the NRC staff for the application of Section 4.15 of the ANSI/IEEE standard:

> Manual switching to the more restrictive setpoint for the APRMs in the reactor protection system is acceptable for BWRs if sufficient administrative controls exist to assure that the more restrictive setpoints are in effect when required by the plant Technical Specifications.

II. EVALUATION AND RECOMMENDATIONS

The current CNS Technical Specifications do not permit single-loop plant operation at reduced power for more than 24 hours, although it is highly desirable from a plant availability/outage planning standpoint. The licensee's proposed Technical Specification changes would allow the reactor to operate at reduced power (not greater than 50%) with one recirculation loop inoperable for more than 24 hours if certain changes are made to the reactor protection systems. Specifically, these changes are to the scram trip setpoints of the Average Power Range Monitor (APRM) and the rod block settings of the Rod Block Monitor (RBM) systems.

Because of the different flow rate and flow path during single-recirculation-loop operation, the APRM SCRAM trip settings, which are flow-biased according to the equation in the proposed technical specifications, require resetting 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 core protection for a postulated rod withdrawal error.

The licensee provided the following technical specification bases for the APRM SCRAM trip settings:

> The average power range monitoring (APRM) system, which is calibrated using heat balance data taken during steady state conditions, reads in percent of rated thermal power (2381 MWt). Because fission chambers provide the basic input signals, the APRM system responds directly to average neutron flux. During transients, the instantaneous rate of heat transfer from the fuel (reactor thermal power) is less than the instantaneous neutron flux due to the time constant of the fuel. Therefore, during abnormal operational transients, the thermal power of the fuel will be less than that indicated by the neutron flux at the scram setting. Analyses demonstrate that with a 120 percent scram trip setting, none of the abnormal operational transients analyzed violate the fuel Safety Limit and there is a substantial margin from fuel damage. Therefore, the use of flow-referenced scram trip provides even additional margin.
>
> An increase in the APRM SCRAM trip setting would decrease the margin present before the fuel cladding integrity Safety Limit is reached. The APRM SCRAM trip setting was determined by an analysis of margins required to provide a reasonable range for maneuvering during operation. Reducing this operating margin would increase the frequency of spurious scrams which have an adverse effect on reactor safety because of the resulting thermal stresses. Thus, the APRM SCRAM trip setting was selected because it provides adequate margin for the fuel cladding integrity Safety Limit yet allows an operating margin that reduces the possibility of unnecessary scrams.
>
> The scram trip setting must be adjusted to ensure that the LHGR (Linear Heat Generation Rate) transient peak is not increased for any combination of maximum fraction of limiting power density (MFLPD) and reactor core

thermal power. The scram setting is adjusted in accordance with the formula in Specification 2.1.a.1.a, when the MFLPD is greater than the fraction of rated power (FRP). This adjustment may be accomplished by increasing the APRM gain, and thus reducing the slope and intercept point of the flow-referenced APRM High Flux scram Curve by the reciprocal of the APRM gain change.

Analyses of the limiting transients show that no scram adjustment is required to assure MCPR (Minimum Critical Power Ratio) above the safety limit when the transient is initiated from the operating MCPR limit.

The licensee provided the following technical specification bases rod-block trip settings:

Reactor power level may be varied by moving control rods or by varying the recirculation flow rate. The APRM system provides a control rod block which is dependent on recirculation flow rate to limit rod withdrawal; thus protecting against a MCPR of less than the MCPR fuel cladding integrity safety limit. The flow variable trip setting provides substantial margin from fuel damage, assuming a steady state operation at the trip setting, over the entire recirculation flow range. The margin to the Safety Limit increases as the flow decreases for the specified trip setting versus flow relationship; therefore the worst case MCPR which could occur during steady state operation is at 108% of rated thermal power because of the APRM rod block trip setting. The actual power distribution in the core is established by specified control rod sequences and is monitored continuously by the in-core LPRM system. As with the APRM SCRAM trip setting, the APRM rod block trip setting is adjusted downward if the maximum fraction of limiting power density exceeds the fraction of rated power; thus preserving the APRM rod block safety margin. As with the scram setting, this may be accomplished by adjusting the APRM gain.

The licensee indicated in reference 6 that CNS Procedure 10.1 :tled "APRM Calibration" was recently modified to include a provision APRM gain adjustment to account for the difference between effective /e flow for single-loop and two-loop operation. This modification adds term 0.66 ΔW to the APRM readings. After completion of the APRM ıstment, the results are reviewed by the Shift Supervisor and the CNS :neering Department. This procedure ensures the necessary adjustments performed properly, and is consistent with Section 4.15 of IEEE Std ·1971.

The manual APRM gain adjustment to accommodate single loop operation is the only change imposed upon the CNS reactor protection system (RPS). This change will not cause the RPS to violate General Design Criteria 20 to 24 of 10CFR50 Appendix A.

The Stability Analysis section of NEDO-24258 indicates that the least stable power/flow conditions attainable under normal conditions occur at natural circulation with the control rods set for rated power and flow. This condition may be reached following the trip of both recirculation pumps. One pump running at minimum speed is more stable than operating with natural flow only, but is less stable than operating with both pumps operating at minimum speed. Under single-recirculation-loop operation, the flow control should be in manual, since control oscillations may occur in the recirculation flow control system under these conditions. We recommend that the licensee revise the technical specifications to include the requirement of manual control of recirculation flow by the operator, as opposed to automatic control during single-recirculation-loop operation.

Because of the different flow pattern during single-recirculation-loop operation, 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 than accurate, but some others will be erroneous. All anomalous control room indications must be corrected or warning-tagged for the duration of the single-recirculation-loop operation, as required by section 4.20 of IEEE Std-279-1971.

III. CONCLUSIONS

Based on our review of the information and documents provided by the licensee, we conclude that the more conservative setpoints for the APRM and RBM will be properly adjusted to protect the reactor for single-recirculation-loop operation.

The current manual method of setting the APRM and RBM trip points meets IEEE Std 279-1971, and is acceptable for Cooper Nuclear Station.

The proposed changes to the reactor protection system (RPS) to accommodate single loop operation at CNS do not cause the RPS to violate General Design Criteria 20 through 24 of 10CFR50 Appendix A, and are considered acceptable.

In order to prevent the potential loss of LPCI, we recommend that the licensee revise the technical specifications to include the requirements of proper valve alignment and tagging prior to commencement of single-recirculation-loop operation.

In order to achieve stable recirculation flow control during single-recirculation-loop operation, we recommend that the licensee revise the technical specifications to include the requirement of manual control of recirculation flow by the operator, as opposed to automatic control during single-recirculation-loop operation.

All anomalous control room indications must either be corrected for single-recirculation-loop operation or warning-tagged.

We conclude that upon successful implementation of the above recommended actions, the proposed licensee ammendment for single-recirculation-loop operation at Cooper Nuclear Station is acceptable.

REFERENCES

1. Nebraska Public Power District letter (J. M. Pilant) to NRC (T. A. Ippolito), "Change to Appendix A Technical Specifications-Single Loop Operation," dated August 5, 1980.

2. General Electric Company, Nuclear Energy Division, "Cooper Nuclear Station Single Loop Operation," NEDO-24258, May 1980.

3. Nebraska Public Power District letter (J. M. Pilant) to NRC (D. B. Vassallo), "Single Loop Operation-Response to NRC Questions," dated May 6, 1982.

4. Telephone conference call, NRC (R. Clark, B. Siegel); NPPD (J. Weaver); EG&G San Ramon (D. Laudenbach), July 21, 1982.

5. Telephone conference call, NRC (R. Clark, B. Siegel, J. T. Beard); NPPD (J. Weaver); EG&G San Ramon (D. Laudenbach), July 22, 1982.

6. Nebraska Public Power District letter (J. M. Pilant) to NRC (D. B. Vassallo), "Single Loop Operation Revised Technical Specifications," dated July 28, 1982.

7. Code of Federal Regulations, Title 10, Part 50, Appendix A, General Design Criteria for Nuclear Power Plants, 1981.

8. IEEE Std-279-1971: Criteria for Protection Systems for Nuclear Power Generating Stations, dated 1971.

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

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VASSALLO

FEB 2 1984

MEMORANDUM FOR: G. C. Lainas, Assistant Director
for Operating Reactors, DL

FROM: L. S. Rubenstein, Assistant Director
for Core and Plant Systems, DSI

SUBJECT: SLO OPERATION SER'S

As discussed in our meeting of January 26, we are withdrawing our SER approvals for all the plants currently requesting permanent SLO. This decision is based on new data which indicates the potential for local thermal hydraulic instabilities which would not be detected by only monitoring APRM flux noise, which we previously recommended.

We are continuing to evaluate this problem and expect to establish criteria for acceptable SLO in the near future.

Alan La Pa for
L. S. Rubenstein, Assistant Director
for Core and Plant Systems, DSI

cc: R. Mattson
D. Eisenhut
L. Phillips
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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

NOV 14 1979

MEMORANDUM FOR: T. Ippolito, Chief
Operating Reactors Branch #3
Division of Operating Reactors

FROM: G. Lainas, Chief
Plant Systems Branch
Division of Operating Reactors

SUBJECT: MONTICELLO NUCLEAR GENERATING PLANT, UNIT 1 - SAFETY EVALUATION FOR N-1 LOOP OPERATION

In response to technical assistance request, TAC 6190, enclosed is the Plant Systems Branch Safety Evaluation Report for single loop operation of Monticello, Unit 1. We find the proposed modifications to the plant for single loop operation as described in Northern States Power Company's submittal to be acceptable.

G. Lainas, Chief
Plant Systems Branch
Division of Operating Reactors

Enclosure:
Safety Evaluation Report

Contact:
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ENCLOSURE

SAFETY EVALUATION REPORT

N-1 LOOP OPERATION

MONTICELLO NUCLEAR GENERATING PLANT UNIT 1

I. INTRODUCTION

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By letter to the U. S. Nuclear Regulatory Commission (NRC) dated September 7, 19~~79,~~ the Northern States Power Company (NSP) 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, dated June 1976, to ensure that its generic analyses for boiling water reactors (BWR) 3/4 were applicable to the Monticello Nuclear Generating Plant, Unit 1. GE-NED submitted an additional report (NEDO-20566-2, dated July 1978) of an analytical model for a Loss-of Coolant-Accident (LOCA) with one recirculation loop out-of-service which is presently 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 using the following criteria: IEEE Std. 279-1971; the Code of Federal Regulations, Title 10, Part 50.46; and Title 10, Part 50, Appendix A and Appendix K.

II. EVALUATION

The enclosed technical evaluation was prepared for us by Lawrence Livermore Laboratory/EG&G as part of our technical assistance program.

III. CONCLUSION

The consultant has reviewed Northern States Power Company's submittal for license amendment for single-loop operation at the Monticello Nuclear Generating Plant, Unit 1, and concluded

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that the modifications satisfy the IEEE Std. 279-1971 criteria and are acceptable. The submittal was based on the analysis in NEDO-21252 performed by the nuclear steam supply system manufacturer (GE-NED). The manufacturer had, however, not analyzed the performance of the Emergency Core Cooling System (ECCS) during single-loop operating conditions.

A new analysis has been performed by GE-NED for a LOCA with one recirculation loop out-of-service. This analysis, reported in NEDO-20566-2, includes the ECCS single-loop analysis and was provided to satisfy the Code of Federal Regulations, Title 10, Part 50, Appendix K.

The consultant also concluded that if an additional review of the EI&C design aspects is required as part of the staff's review of NEDO-20566-2, the licensee will be required to update its submittal based on that new analysis. Such a review, if required, will then be presented as a supplement to the consultant's technical evaluation.

Based on our review of consultant's technical evaluation, we conclude that conceptional design as presented in the licensee submittal and reviewed in the consultant's technical evaluation is acceptable. However, the licensee's submittal did not include a design of hard-wire modifications (see Section 2.2 of attached technical evaluation) to the reactor protection system that will enable the operator to make setpoint changes from the front of the nuclear instrument cabinet. It is, therefore, concluded that before operation in the single-loop mode can be implemented at Monticello, Unit 1. The licensee must accomplish the aforementioned modifications to the reactor protection system in a manner that satisfies IEEE Stds. 279-1971, 323-1971, and 344-1975.

TECHNICAL EVALUATION OF THE

ELECTRICAL, INSTRUMENTATION, AND CONTROL DESIGN ASPECTS

OF

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THE MONTICELLO NUCLEAR GENERATING PLANT, UNIT 1

(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) 3/4 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 resetting 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 I2.[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^{o}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

We conclude that the, Northern States Power Company's license amendment submittal for single-loop operation of the Monticello nuclear generating plant, Unit 1, satisfies the IEEE Std-279-1971 criteria and is acceptable. The submittal was based on the analysis in NEDO-21252[2] performed by the nuclear steam supply system manufacturer (GE-NED). The manufacturer had, however, not analyzed the performance of the emergency core cooling system (ECCS) during single-loop operating conditions.

A new analysis has been performed by GE-NED for a LOCA with one recirculation loop out-of-service. This analysis, reported in NEDO 20566-2,[3] includes the ECCS single-loop analysis and is in accordance with the Code of Federal Regulations, Title 10, Part 50, Appendix K.[7]

If an additional review of the EI&C design aspects is required as a result of NEDO-20566-2,[3] the licensee will be required to update its submittal based on that new analysis. The review will then be presented as a supplement to this technical evaluation.

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.).