August 1982

DRAFT 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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EG&G, Inc.
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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 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 a request for additional information, 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] concerning protection system trip point setting changes for CNS single-loop operation, and documented by the licensee's letter dated July 28, 1982 [Ref. 6].

The purpose of this report is to evaluate 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 IEEE Std-279-1971 [Ref. 7]; NRC Branch Technical Position EICSB-12 [Ref. 8]; the Code of Federal Regulations, Title 10, Part 50, Appendix A, "General Design Criteria for Nuclear Power Plants" [Ref. 9]; and NRC Review Criteria detailed in Project 8 of FIN/189A No. A-0250 [Ref. 10].

II. EVALUATION AND RECOMMENDATIONS

The licensee indicated that current CNS technical specifications do not allow plant operation beyond a relatively short period of time if an idle recirculation loop cannot be returned to service. In the event maintenance of a recirculation pump or other component renders one loop inoperable, the capability of operating at reduced power with a single-recirculation-loop is highly desirable from a plant availability/outage planning standpoint.

The licensee's proposed technical specifications would allow the reactor to operate in single-recirculation-loop operation for 24 hours before making any setpoint changes to the reactor protection system. With one recirculation loop out of service for greater than 24 hours, the reactor would not be operated at a rated thermal power greater than 50%. In order to continue to operate the reactor in single-recirculation-loop operation beyond 24 hours, it will be necessary to make setpoint changes to the SCRAM trip settings of the average power range monitor (APRM) system and to the rod-block settings of the rod block monitor (RBM) system. Because of the different flow rate and 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.
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> 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 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 above the safety limit when the transient is initiated from the operating MCPR limit.

The licensee provided the following technical specification bases for 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 IPRM 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 entitled "APRM Calibration" was recently modified to include a provision for APRM gain adjustment to account for the difference between effective drive flow for single-loop and two-loop operation. This modification involves adding the term $0.66\Delta W$ to the APRM readings. After completion of the APRM adjustment, the results are reviewed by the Shift Supervisor and the CNS Engineering Department. This procedure ensures the necessary adjustments are performed properly. Because sustained single-recirculation-loop operation is a rare event and not a planned mode of operation, we find the above manual trip point setting procedures to be in accordance with the requirements of Section B.2 of BTP EICSB-12 and acceptable for an interim period up to the next plant refueling. However, in order to satisfy all review criteria, we recommend that the single-recirculation-loop APRM and RBM trip set points be made automatic or hard-wired (switch-selectable) from the control panel. The changes should be made at the first opportunity and installed no later than the next CNS refueling outage. These hardwire modifications must be submitted to the staff for review prior to installation.

The GE-NED report NEDO-24258 safety analyses were performed assuming the recirculation equalizer valves were closed. Further, the report indicated that the discharge valve in the idle recirculation loop is normally closed. However, if its closure is prevented, the suction valve in the loop should be closed to prevent the loss of Low Pressure Coolant Injection (LPCI) flow out of a postulated break in the idle loop suction line. We recommend that the licensee revise the proposed technical specifications to include the requirement for proper valve alignment and tagging prior to commencement to single-recirculation-loop operation.

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 master manual, since control oscillations may occur in the recirculation flow control system under these conditions. We recommend that the licensee revise the proposed technical specifications to include the requirement of master 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 is acceptable for an interim period of up to the next plant refueling outage. In order to satisfy the requirements of the review criteria, it will be necessary that these trip point settings be made automatic or hardwired (switch-selectable) from the control panel. The hardwire modifications will require staff review prior to installation.

In order to prevent the potential loss of LPCI, we recommend that the licensee revise the proposed 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 proposed technical specifications to include the requirement of master 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. IEEE Std-279-1971: Criteria for Protection Systems for Nuclear Power Generating Stations, dated 1971.

8. NRC Branch Technical Position EICSB-12, "Protection System Trip Point Changes for Operation with Reactor Coolant Pumps Out of Service," dated November 24, 1975.

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

10. Project and Budget Proposal for NRC Work, NRC Form 189.

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

SEP 2 0 1982

MEMORANDUM FOR: Themis P. Speis, Assistant Director for Reactor Safety
Division of Systems Integration

FROM: Brian W. Sheron, Chief, Reactor Systems Branch, DSI

SUBJECT: BWR SINGLE LOOP OPERATION - STATUS REPORT #2

Under multiplant action item E-04, there are 15 plants which involve single loop operation (SLO) the status of each plant is given below.

Plant Name	SER Issued for 50% Operation	Licensing Amendment Issued	Notes
1. Dresden-2	July 9, 1981	Yes	
2. Dresden-3	July 9, 1981	Yes	
3. Quad Cities-1	July 9, 1981	Yes	
4. Quad Cities-2	July 9, 1981	Yes	
5. Peach Bottom-1	May 15, 1981	Yes	
6. Peach Bottom-2	May 15, 1981	Yes	
7. Duane Arnold	November 19, 1981	No	1
8. Cooper	December 10, 1981	No	1
9. Pilgrim-1	December 15, 1981	No	1
10. Browns Ferry-1	August 16, 1982	No	1,2
11. Browns Ferry-2	August 16, 1982	No	1,2
12. Browns Ferry-3	August 16, 1982	No	1,2
13. Monticello	September 10, 1982	No	1
14. Brunswick-1	Being Reviewed, SER Due on November 1, 1982		
15. Brunswick-2	Being Reviewed, SER Due on November 1, 1982		

Notes
1. Instrumention & Control review by EG&G, San Ramon Operations, CA is not completed. Licensing amendment will be issued after we get an acceptable TER from EG&G.

2. CPB sections on thermal hydraulics and stability analysis are not included in the SER submitted for Browns Ferry Units 1,2,&3.

As a follow up to the Browns Ferry-1 meeting with GE on single loop operation, questions were forwarded to T. Novak by memo dated January 15, 1982 from T. Speis for sending to GE and the utility group. Recently, approval was granted by GAO for sending the questions to the licensees. Questions are expected to be sent to the individual

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licensees during first week of October 1982. Approval for single loop operation at a power greater than 50% can be granted only after staff concerns stemming from Browns Ferry- Unit 1 single loop operation are satisfied.

Brian W. Sheron

Brian W. Sheron, Chief
Reactor Systems Branch, DSI

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