1183-4144

TECHNICAL EVALUATION OF THE ELECTRICAL, INSTRUMENTATION, AND CONTROL DESIGN ASPECTS OF THE PROPOSED LICENSE AMENDMENT FOR SINGLE-LOOP OPERATION OF THE BRUNSWICK STEAM ELECTRIC PLANT, UNIT 2

(DOCKET NO. 50-324)

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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

James H. Cooper

Approved for Publication

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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 for the Brunswick steam electric plant, Unit 2. 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.

The U. S. Nuclear Regulatory Commission funded the work under the authorization entitled "Electrical, Instrumentation and Control System Support," B&R 20 19 04 031, FIN A-0231.

This work was performed by EG&G, Inc., Energy Measurements Group, San Ramon Operations, for Lawrence Livermore Laboratory under U. S. Department of Energy contract number DE-AC08-76NV01183.

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THE BRUNSWICK STEAM ELECTRIC PLANT, UNIT 2

(Docket No. 50-324)

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 3, 1976, the Carolina Power & Light Company (CP&L) submitted information to support its proposed license amendment to operate the Brunswick steam electric plant, Unit 2, 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-21281,[2] dated May 1976, to ensure that its generic analyses for boiling water reactors (BWR 3 and/or 4) were applicable to the Brunswick steam electric plant, Unit 2. 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 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-21281[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-21281[2] the only changes necessary to the reactor protection system (RPS) for single-loop operation, are:

(1) Modifications to the rod-block setpoints of the rod-block monitor (RBM) system.

(2) Modifications to the SCRAM trip settings of the average power range monitor (APRM) system.

(3) Reduction of 0.82 in the maximum average planar linear heat generation rate (MAPHLGR) limit for the fuel.

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 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-21281[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 improved in the manual mode. Manual control is assumed in NEDO-21281.[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-21281[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-21281[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 Carolina Power and Light 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. CP&L Letter to NRC (B. Rusche), dated September 3, 1976.

2. General Electric Company, Nuclear Energy Division, Brunswick Steam Electric Plant Unit 2 License Amendment Submittal for Single-Loop Operation With the Bypass Flow Holes Plugged and with LPSI Modification, NEDO-21281 (May 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 (n. d.).

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. CP&L, Final Safety Analysis Report for Brunswick Steam Electric Plant (FSAR) (n. d.).

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

Docket Nos. 50-325
and 50-324

August 24, 1981

Mr. J. A. Jones
Senior Executive Vice President
Carolina Power & Light Company
336 Fayetteville Street
Raleigh, North Carolina 27602

RE: BRUNSWICK STEAM ELECTRIC PLANT UNITS 1 AND 2

Dear Mr. Jones:

On July 2, 1981, we sent a letter to all licensees who have requested approval to operate on a continuing basis at power levels above 50% with only one recirculation loop in the event the other loop is inoperative. You and other BWR licensees received a copy of one of these letters since we expect most BWR facilities would like to have this flexibility. In the letter we proposed a meeting to obtain a better understanding of what might have caused variations in jet pump flow and related parameters at Browns Ferry Unit No. 1 during single loop operation and how this incident should affect approval of single loop operation at other facilities.

You have indicated to your NRC project manager that you are interested in attending the proposed meeting. The meeting will be held at 9:00 A.M., Wednesday, September 9, 1981 in room P-118, Phillips Building, 7920 Norfolk Avenue, Bethesda, Maryland. You are requested to advise your project manager of the people who will be attending this meeting from your organization.

Sincerely,

T. Ippolito

Thomas A. Ippolito, Chief
Operating Reactors Branch #2
Division of Licensing

Enclosure:
Meeting Agenda

cc w/enclosure
See next page

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Proposed Meeting with BWR Applicants and Licensees on Single Loop Operation

Purpose of Meeting:

1. To determine what may have caused the jet pump flow and other variations experienced by Browns Ferry Unit 1 during single loop operation and

2. Evaluate whether the Browns Ferry experience should result in power limits for other BWRs operating on a single loop.

Agenda:

1. Discussion of what may have caused the unexpected variations in operating parameters when Browns Ferry Unit 1 exceeded about 60 percent rated power while operating with only one recirculation loop.

2. Discussion of parameters affected at Browns Ferry 1 (i.e., jet pump flow, neutron flux, core flow, core pressure drop, etc.)

3. Discussion of whether the Browns Ferry 1 experience would be expected at other BWRs operating on one recirculation loop. If ~~so,~~ are safety limits likely to be violated or cause complications with respect to core stability, core flow symmetry, pump cavitation or damage to the jet pumps and reactor vessel internals.

4. Discussion of the benefits vs. potential problems and cost of testing single loop operation in another BWR that is instrumented to detect what paramaters are affected.

5. Evaluation of whether single loop operation at power levels about 50 to 55 percent is a safe and prudent means of reactor operation.

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

September 23, 1981

Docket Nos. 50-325
and 50-324

Mr. J. A. Jones
Senior Executive Vice President
Carolina Power & Light Company
336 Fayetteville Street
Raleigh, North Carolina 27602

RE: BRUNSWICK STEAM ELECTRIC PLANT UNITS 1 AND 2

Dear Mr. Jones:

My letter to you of August 24, 1981 informed you that we were proposing a meeting with licensees who have requested approval to operate on a single recirculation loop. The purpose of the meeting is to determine what may have caused variations in jet pump flow at Browns Ferry Unit No. 1 while operating on a single loop and what impact this should have on approval of other facilities to operate on one loop. Licensees and applicants who have not requested approval for single loop operation were also invited to the meeting.

As you were advised by your licensing project manager, the meeting scheduled for September 9, 1981 had to be postponed to allow more time for analysis of relevant operating data. We apologize for this inconvenience. The meeting will be held at 9:00 AM on Thursday, October 22, 1981 in Room P-118, Phillips Building, 7920 Norfolk Avenue, Bethesda, Maryland. It would be appreciated if you would inform your licensing project manager of the number of people who will be attending this meeting from your organization.

Sincerely,

T. A. Ippolito

Thomas A. Ippolito, Chief
Operating Reactors Branch #2
Division of Licensing

cc: See Next Page

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Mr. J. A. Jones

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UNITED STATES
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WASHINGTON, D. C. 20555

2-6

Docket No. 50-324

October 9, 1981

Mr. J. A. Jones
Senior Executive Vice President
Carolina Power & Light Company
336 Fayetteville Street
Raleigh, North Carolina 27602

Dear Mr. Jones:

RE: BRUNSWICK STEAM ELECTRIC PLANT UNIT NO. 2

We have reviewed your single loop license amendment request dated August 12, 1981. In that request you assert that the potential effects of a pump seizure accident are bounded by the effects of a Loss of Coolant Accident (LOCA). However, the acceptance criteria for a pump seizure are more stringent than the criteria for a LOCA. Standard Review Plan 15.3.3 (Reactor Coolant Pump Rotor Seizure and Reactor Coolant Pump Shaft Break) requires that for a pump seizure accident any activity release must be such that the calculated doses at the site boundary are a small fraction of the 10 CFR Part 100 guidelines. This is significantly more limiting than the corresponding LOCA criteria. We request therefore that you submit an analysis that demonstrates the acceptability of a pump seizure accident during single loop operation. We will not be able to complete action on your request until we have received and reviewed this additional information.

Sincerely,

T. A. Ippolito

Thomas A. Ippolito, Chief
Operating Reactors Branch #2
Division of Licensing

cc: See Next Page

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