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

January 22, 1990

Docket No. 50-416

LICENSEE: System Energy Resources, Inc. (SERI)

FACILITY: Grand Gulf Nuclear Station, Unit 1 (GGNS-1)

SUBJECT: SUMMARY OF JANUARY 4, 1990 MEETING REGARDING CYCLE 5 LICENSING ACTIVITIES

The NRC staff met with the licensee and representatives of Advanced Nuclear Fuels, Inc. (ANF) at the NRC office in Rockville, Maryland to discuss cycle 5 licensing activities. Enclosure 1 is a list of participants in the meeting. Enclosure 2 is a copy of the handout prepared by SERI which includes the meeting agenda. Enclosure 3 is a staff Safety Evaluation for Clinton with feedwater heaters out of service.

The cycle 5 reload will be the first full reload at GGNS-1 using ANF's 9x9-5 fuel. Energy characteristics are approximately equal to cycle 4 and there are slight increases in designed discharge burnup and enrichment. Major design features of the fuel were described by the licensee.

The NRC Staff discussed the status of review of analytical methodologies used by the licensee for the cycle 5 core design and stated that the reviews of two of the methodologies, which are being performed by a contractor, are being delayed. SERI plans to submit a plant specific reload report using these methodologies. The NRC staff stated that this would be appropriate even though the generic reviews are not final.

The licensee briefly described the scope of neutronics, transient, loss of coolant, thermal limits, mechanical and rod drop analyses. Criticality analysis for the fuel storage racks was also discussed. Stability analysis and confirmation of the cycle 5 core design was discussed at some length; SERI expects to see only minor changes in the degree of stability from the previous core.

The licensee presented a summary of submittal dates and a projected schedule for licensing submittals required for the cycle 5 reload. Currently the licensee plans to make the cycle 5 reload submittal in July 1990 to support the October through November 1990 outage schedule. The NRC staff expressed concerns that a July submittal may not allow adequate time for review due to resources considerations. The NRC staff recommended that SERI plan to make their submittal earlier.

The NRC staff noted that recent changes to the Updated Final Safety Analysis Report (UFSAR) had added transient and accident analyses for feedwater heaters out-of-service and questioned whether the staff had reviewed these

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analyses. (UFSAR, Section 15 and Appendix 15B). The licensee thought they had been reviewed by the staff in one of the reload applications, probably the one where the maximum extended operating domain was approved. Subsequent to the meeting, the staff found that it had not reviewed accident analyses for GGNS-1 but that it had reviewed and approved an analysis for Clinton with feedwater heaters out-of-service and decreased feedwater temperature up to 100°F. The staff Safety Evaluation of the Clinton analyses is enclosed for your information. By telephone on January 16, 1990, the staff advised the licensee that analyses for operation with feedwater heaters out-of-service should be included in the next reload application, if such operation is expected to be used.

Original Signed By:

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DATE	:1/22/90	1/22/90	1/22/90	:	:	:	: : :

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ENCLOSURE 1

January 4, 1990 NRC-SERI Meeting

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P. Balmain	NRC/Region II
Y. Balas	SERI
I. Nir	SERI
F. Smith	SERI
P. Brown	SERI
B. Copeland	ANF
T. Krysiuski	ANF

ENCLOSURE 2

NRC/SERI MEETING

GRAND GULF CYCLE 5 LICENSING ACTIVITIES

JANUARY 4, 1990

AGENDA

I. INTRODUCTION

A) MEETING OBJECTIVES & AGENDA

B) SCHEDULE

II. CYCLE 5 DESIGN SUMMARY

A) CYCLE 5 OVERVIEW

B) 9x9-5 RELOAD FUEL

C) TECH SPECS IMPACT

D) SAFETY ANALYSIS

III. CYCLE 5 SAFETY ANALYSES

A) NEW METHODOLOGIES

- CASMO/MICROBURN-B
- COTRANSA 2
- ANFB
- REVISED METHODOLOGY FOR SAFETY LIMITS

B) ANALYSES AND SCOPE

- NEUTRONICS
- TRANSIENTS
- LOCA
- THERMAL LIMITS
- MECHANICAL DESIGN

C) CRITICALITY

D) STABILITY

E) LICENSING IMPACT

F) STATUS OF ANALYSES AND TOPICALS

IV. SCHEDULE

A) ANF GENERIC AND PLANT SPECIFIC DOCUMENTS

B) RELOAD SUBMITTAL

C) GENERAL SCHEDULE

V. WRAP-UP/SUMMARY

PURPOSE OF MEETING

INTRODUCTION OF NEW PERSONNEL

UPDATE THE NRC ON CURRENT PLANT STATUS AND CYCLE 5 SCHEDULE

INFORM THE NRC OF THE CHANGES PLANNED FOR CYCLE 5

REVIEW THE SUBMITTALS NEEDED TO SUPPORT CYCLE 5 CHANGES AND THE SCHEDULES NECESSARY TO SUPPORT CYCLE 5 STARTUP DATE

CONFIRM STAFF'S SUPPORT TO MEET STARTUP SCHEDULE

SCHEDULE - KEY DATES

CYCLE 5 FUEL RECEIPT AT SITE	AUGUST 1990
EOC 4	OCTOBER 1990
BOC 5	NOVEMBER 1990

CYCLE 5 OVERVIEW

-	FUEL TYPE	FIRST FULL RELOAD 9x9-5
-	ENERGY	APPROXIMATELY EQUAL TO CYCLE 4
-	DESIGN DISCHARGE BURNUP	INCREASE FROM 34 TO 36 GWD/MT
-	ENRICHMENT	SLIGHT INCREASE

9x9-5 DESCRIPTION

DESIGN FEATURES

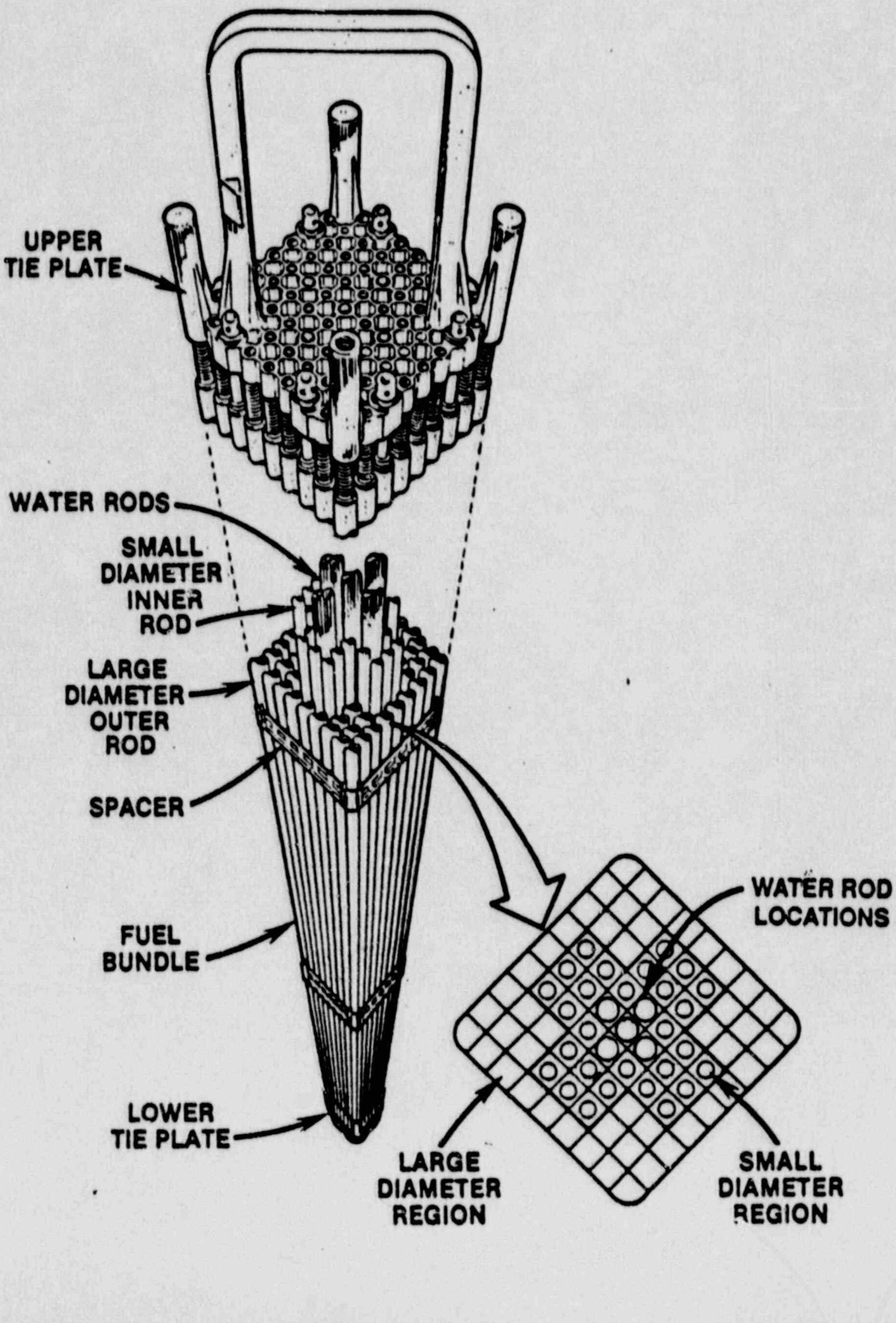
- TWO FUEL ROD DIAMETERS
- FIVE CENTRALLY LOCATED WATER RODS

PERFORMANCE CHANGE RELATIVE TO CURRENT 8x8 EXPERIENCE

- BETTER CRITICAL POWER PERFORMANCE DUE TO MORE EFFECTIVE DISTRIBUTION OF COOLANT
- IMPROVED LOCA PERFORMANCE DUE TO LOWER LHGR, GREATER HEAT TRANSFER AREA
- MORE MANEUVERING FLEXIBILITY ALLOWED DUE TO LOWER LHGR

4 LEAD FUEL ASSEMBLIES OF SIMILAR DESIGN ARE INCLUDED IN CURRENT CYCLE

9x9-5 C-LATTICE



TECH SPEC IMPACT

- CLEANUP
- CYCLE 5 SPECIFIC
- GL 88-16

SAFETY ANALYSIS

- LICENSING CONTINUITY
- NEW METHODOLOGIES

NEW METHODOLOGIES FOR CYCLE 5

CASMO/MICROBURN-B NEUTRONICS CODES (XN-NF-80-19(P), VOL. 1, SUPPLEMENT 3)

COTRANSA2 SYSTEM RESPONSE CODE (ANF-913(P), AND SUPPLEMENTS)

ANFB CHF CORRELATION (ANF-1125(P), AND SUPPLEMENT 1)

REVISED SAFETY LIMIT METHODOLOGY (ANF-524(P), REVISION 2)

SAFETY LIMIT CHANNEL BOW METHODOLOGY SUPPLEMENT (ANF-524(P), REVISION 2, SUPPLEMENT 1)

ANALYSES AND SCOPE

WORKSCOPE FOR CYCLE 5 INCLUDES A REANALYSIS OF THE APPROPRIATE EVENTS CONSISTENT WITH THE CURRENT MEOD.

NEUTRONICS ANALYSIS

- COLD SHUTDOWN MARGIN
- STANDBY LIQUID CONTROL
- LOSS OF FEEDWATER HEATING
- FLOW EXCURSION EVENT
- ROD WITHDRAWAL ERROR
- MISLOADED BUNDLE

ANF PERFORMING THE NEUTRONIC/CYCLE DESIGN ACCORDING TO THE ACCEPTED NEUTRONIC METHODOLOGY BUT USING CASMO/MICROBURN-B INSTEAD OF XFYRE/XTGBWR

TRANSIENT ANALYSIS

- LOAD REJECTION W/O BYPASS
- FEEDWATER CONTROLLER FAILURE
- SINGLE LOOP OPERATION
- OVERPRESSURIZATION

TRANSIENT SYSTEM RESPONSE WILL BE ANALYZED USING THE COTRANSA2 CODE (ANF-913-(P) AND SUPPLEMENTS).

CHF PERFORMANCE WILL BE DETERMINED USING THE EXTENDED ANFB CHF CORRELATION (ANF-1125(P), AND SUPPLEMENT 1).

SAFETY LIMITS WILL BE DETERMINED USING THE REVISED SAFETY LIMIT METHODOLOGY (ANF-524(P), REVISION 2 AND SUPPLEMENT 1).

ANALYSES AND SCOPE
(CONTINUED)

LOCA ANALYSIS

- PERFORM THE LIMITING BREAK HEATUP ANALYSIS TO ESTABLISH THE MAPLHGR VALUES FOR THE 9x9-5.

- REMOVE SINGLE LOOP OPERATION SPECIFIC MAPLHGR LIMITS

ANALYSES WILL USE THE APPROVED EXEM/BWR METHODOLOGY (XN-NF-80-19(A), VOLUME 2, REVISION 1)

THERMAL LIMITS

- $MCPR_F$, $MCPR_P$

- LGHR, $LHGRFAC_F$, $LHGRFAC_P$

- MAPLHGR

MECHANICAL ANALYSIS

THE GENERIC MECHANICAL DESIGN REPORT USES APPROVED CODES AND PERFORMED IN A SIMILAR MANNER AS PREVIOUSLY ACCEPTED MECHANICAL DESIGN REPORTS (ANF-88-152(P)).

ROD DROP

REDUCE BPWS OPERABILITY REQUIREMENTS FROM 20% POWER TO 10% POWER BASED ON BWROG ANALYSIS

CRITICALITY

THE CRITICALITY ANALYSIS FOR THE FUEL STORAGE RACKS WILL USE A BOUNDING BUNDLE DESIGN IN ORDER TO MINIMIZE THE NEED FOR FUTURE SUBMITTALS. THE SPENT FUEL RACKS WILL BE MODELED USING SIMILAR METHODOLOGY TO THAT USED IN SUPPORT OF THE CYCLE 4 RELOAD.

THE ANALYSIS WILL USE THE MAXIMUM REACTIVITY POINT IN THE FUEL ASSEMBLIES' LIFETIME CONSIDERING BURNUP AND BURNABLE POISON. THIS ANALYSIS WILL BE PERFORMED CONSISTENT WITH THE REQUIREMENTS OF ANSI/ANS-57.2 - 1983.

THE ANALYSIS WILL CONSIDER THE EFFECTS OF GAPS IN THE BORAFLEX ABSORBER SHEETS.

THE ANALYSIS WILL BE PERFORMED AND SUBMITTED SEPARATELY FROM THE RELOAD ANALYSIS BECAUSE FUEL RECEIPT IS TYPICALLY SCHEDULED 3 - 4 MONTHS PRIOR TO STARTUP.

STABILITY TECH SPEC CONFIRMATION

CURRENT STABILITY TECH SPEC

CONSISTENT WITH BWROG INTERIM CORRECTIVE ACTIONS (ICA)

SER RECEIVED ON 8/31/89

- APPLIES TO ANF 8 x 8
- REQUIRES REEVALUATION FOR OTHER FUEL TYPES

BASIS FOR ACCEPTABILITY IS THAT ANF 8 x 8 AND GE FUEL/CORE STABILITY PERFORMANCE IS NOT SIGNIFICANTLY DIFFERENT

- FUEL PERFORMANCE SENSITIVITY ANALYSES (RETRAN)
- ANF/GE MIXED CORE PERFORMANCE SENSITIVITY ANALYSES (COTRAN)

PREVIOUS GGNS-1 STABILITY TESTS PROVIDE ADDITIONAL SUPPORT OF MARGIN TO INSTABILITY

CYCLE 5 STABILITY CONFIRMATION APPROACH

BWROG ICA ARE ACCEPTABLE FOR GE AND ANF 8 x 8 FUEL

NRC APPROVED CODE FOR STABILITY ANALYSIS WILL BE USED TO EVALUATE THE RELATIVE IMPACT OF A 9x9-5 RELOAD BATCH ON CORE STABILITY.

- ANALYZE SAME STATE POINTS AS IN PREVIOUS CYCLES FOR CYCLE 5
- CALCULATE DIFFERENCE IN DECAY RATIO FROM CYCLE 4
- ASSESS DIFFERENCE RELATIVE TO EXPECTED DECAY RATIO VARIATIONS FROM CYCLE TO CYCLE

CONFIRM ICA ARE ACCEPTABLE FOR CYCLE 5

EXISTING STABILITY TESTS FOR ANF 9 x 9 FUEL LOADINGS PROVIDE ADDITIONAL SUPPORT OF MARGIN TO INSTABILITY

LICENSING IMPACT

1. MAPFAC WILL BE REPLACED BY LGHR FACTOR
2. REMOVAL OF SLO SPECIFIC MAPLHGR LIMITS
3. REVISE TS LIMITS SPECIFIC TO 9x9-5 AS NEEDED
4. REDUCE MAXIMUM POWER LEVEL FOR BPWS OPERABILITY FROM 20% TO 10%
5. IN CONJUNCTION WITH GL 88-16 THERMAL LIMITS MAY BE TAKEN OUT OF THE TS
6. ANALYSES ARE CURRENTLY BEING PERFORMED USING GENERIC METHODOLOGIES UNDER REVIEW.

LICENSING STATUS OF ANF GENERIC METHODOLOGY

TOPICAL REPORT	STATUS
9x9-5 GENERIC MECHANICAL DESIGN REPORT (ANF-88-152(P))	SUBMITTED IN NOVEMBER 1988, RESPONDED TO NRC QUESTIONS ON DECEMBER 15, 1989.
CASMO/MICROBURN-B NEUTRONICS CODES (XN-NF-80-19(P) VOL. 1, SUP 3)	SUBMITTED IN MARCH 1989.
ANFB CRITICAL POWER CORRELATION AND EXTENSION FOR THE 9x9-5 DESIGN (ANF-1125(P), AND SUPPLEMENT 1)	SUBMITTED BASE CORRELATION IN FEBRUARY 1988. SUBMITTED EXTENSION FOR 9x9-5 DESIGN IN APRIL 1989. RESPONDED TO NRC QUESTIONS ON OCTOBER 23, 1989.
COTRANSA2 CODE DESCRIPTION AND PEACH BOTTOM BENCHMARKS (ANF-913(P), VOL. 1 AND SUPPLEMENTS)	SUBMITTED TO NRC IN MAY 1988. SUBMITTED BENCHMARKS WITH MICROBURN INPUT IN JUNE 1989. RESPONDED TO NRC QUESTIONS IN NOVEMBER 1989.
REVISED SAFETY LIMIT METHODOLOGY (ANF-524(P), REVISION 2)	SUBMITTED TO NRC IN APRIL 1989.
REVISED SAFETY LIMIT CHANNEL BOW METHODOLOGY (ANF-524(P), REVISION 2, SUPPLEMENT 1)	SUBMITTED TO NRC IN NOVEMBER 1989 AT NRC'S REQUEST.

SUBMITTAL SUMMARY AND SCHEDULE

ITEM	SCHEDULED SUBMITTAL DATE
COTRANSA2 SYSTEM RESPONSE CODE (ANF-913(P), AND SUPPLEMENTS)	MAY 1988
ANF 9x9-5 MECHANICAL DESIGN REPORT (ANF-88-152(P))	NOVEMBER 1988
ANF CASMO/MICROBURN-B NEUTRONICS CODE (XN-NF-80-19(P), VOL. 1, SUPPLEMENT 3)	MARCH 1989
ANF CHF CORRELATION FOR 9x9-5 DESIGN (ANF-1125(P) AND SUPPLEMENT 1)	APRIL 1989
ANF REVISED SAFETY LIMIT METHODOLOGY (ANF-524(P), REVISION 2)	APRIL 1989
NRC/SERI CYCLE 5 PLANNING MEETING	MAY 1989
COTRANSA2 PEACH BOTTOM BENCHMARKS USING MICROBURN NEUTRONICS	JUNE 1989
SAFETY LIMIT CHANNEL BOW SUPPLEMENT (ANF-524(P), REVISION 2, SUPPLEMENT 1)	NOVEMBER 1989
GENERIC SERs ISSUED	DECEMBER 1989
NRC/SERI CYCLE 5 RELOAD MEETING	JANUARY 1990
CRITICALITY ANALYSIS SUBMITTED	APRIL 1990
CRITICALITY SER ISSUED	JULY 1990
CYCLE 5 RELOAD SUBMITTAL	JULY 1990
FUEL RECEIPT AT GRAND GULF	AUGUST 1990
OUTAGE BEGINS	OCTOBER 1990
CYCLE 5 LICENSE RECEIVED	OCTOBER 1990
CYCLE 5 STARTUP	NOVEMBER 1990

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

ENCLOSURE 3

May 15, 1989

Docket No. 50-461

Mr. Dale L. Holtzscher
Acting Manager - Licensing and Safety
Clinton Power Station
P. O. Box 678
Mail Code V920
Clinton, Illinois 61727

Dear Mr. Holtzscher:

SUBJECT: CLINTON POWER STATION, LICENSE CONDITION 4, CONTROL SYSTEMS FAILURE (TAC NO. 62991)

License Condition 4 of facility operating license NPF-62 for the Clinton Power Station (CPS) required Illinois Power Company (IP) to submit the results of an additional evaluation of control system failures and propose implementations of any corrective actions. By letter dated November 18, 1988, IP submitted the required analysis. Methodology for this analysis was approved in NUREG-0853, Supplement 6, and the staff provided guidelines for certain aspects of the analysis in our request for additional information. Our review found that the analysis followed staff guidelines and approved methodologies, and is, therefore, acceptable. IP has committed to several improvements to minimize the probability of loss of feedwater heating and an administrative procedure calling for reactor shutdown should a loss of feedwater heating result in a feedwater temperature reduction approaching 100°F. The staff finds these improvements to be acceptable. Therefore, the staff considers License Condition 4 to have been satisfied. Should you wish to have License Condition 4 deleted from your operating license, you may request that your license be amended.

Sincerely,

John B. Hickman

John B. Hickman, Project Manager
Project Directorate III-2
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cc: See next page

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

SAFETY EVALUATION REPORT

CONTROL SYSTEM FAILURE REANALYSIS - LICENSE CONDITION 4

ILLINOIS POWER COMPANY

CLINTON POWER STATION

DOCKET NO. 50-461

1.0 INTRODUCTION

The Clinton Safety Evaluation Report (SER) outstanding issue number 15 deals with multiple control system failure resulting from high-energy line breaks, common power source failure or sensor malfunction. The staff concern was that the subject control system failure would result in more serious consequences than those analyzed in Chapter 15 of Clinton's FSAR. The staff requested that the applicant identify those sources, which provide power to two or more control systems and demonstrate that failures of these power sources will not result in consequences outside the bounds of the FSAR Chapter 15 analyses. In addition, the applicant was asked to review the designs to determine whether harsh environments associated with high-energy line breaks (HELBs) might cause control system malfunctions resulting in consequences more severe than those analyzed in FSAR Chapter 15. IP's response (analysis) did not consider the effects of all nonsafety-related control system failures for each FSAR Chapter 15 event. In response to the staff's request for additional information, the licensee proposed a complete re-review of the control system failure analysis and a submittal of "Qualitative Event Analysis" to address the staff's concerns and questions. The licensee's proposal was found acceptable in Section 7.7.3.1 of NUREG-0853, Supplement 6 (Reference 1) and was made a licensing condition for Clinton full power operation by NRC letter to IP dated April 17, 1987 (Reference 2). By letter dated November 18, 1988 (Reference 3), IPCO submitted the required analysis. The submittal consisted of a "Combinatory Qualitative Event Analysis," licensee's answer to the six NRC questions, and a proprietary quantitative analysis of a special transient event by General Electric.

The scope of the licensee's analysis was defined in letters dated April 17, May 15 and July 16, 1986 (Refs. 4-6). Additional information was submitted on March 20, 1989 (Ref. 7). The worst case event identified is the loss of feedwater heating with turbine trip and main steam turbine bypass failure. The submittal included a General Electric transient analysis (Ref. 8) which assumed a 100°F loss of feedwater temperature which showed no fuel damage. However, actual Clinton Station operating experience showed that a feedwater temperature drop greater than 100°F could occur.

The licensee analyses included the effect of a single active failure in a mitigating safety system to assure that a sufficient number of such systems will be available for accident mitigation. In addition, the licensee

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committed to several improvements to minimize the probability of loss of feedwater heating and to instituting several operating procedure changes to either prevent or effectively mitigate feedwater losses.

2.0 REACTOR SYSTEMS EVALUATION

The approach taken in the reanalysis was to attempt to identify the non-safety control systems which could affect the reactor. All failure modes of these systems were identified and assessed for event sequences that may not be bounded by the existing FSAR Chapter 15 analysis. The worst case identified is the loss of feedwater heating with turbine trip and failure of the turbine bypass system. An analysis of this event by GE showed that fuel damage would not occur if the feedwater temperature decrease is limited to 100°F. However, Clinton has experienced a loss of feedwater heating with a temperature drop greater than 100°F (Ref. 9).

The submitted study was carried out by QUADREX, a licensee consultant. The object of the analysis is to determine whether the consequences of multiple control system failures are bounded by the Clinton FSAR Chapter 15 events and whether the failures would have an adverse effect on the ability to achieve plant cold shutdown conditions. The methodology assumed that all combinations of non-safety related control system failures are considered likely to occur, regardless of power source, common instrument sensor, or proximity to a high energy line. The Chapter 15 events were not modified, rather, they were considered initiating events that were examined for potential exacerbation by non-safety control system failures. However, systems comprised of structures alone or information systems that merely provide alarms, annunciations, or information to the control room operators were not considered. In addition, systems whose failures would not affect reactor parameters or influence plant operation were eliminated from further analysis. Thus, the systems combinations examined whose failure could affect reactor parameters are:

- loss of feedwater heating combined with non-safety related control system failures
- feedwater controller failure combined with non-safety related control system failures
- turbine pressure regulator failure combined with non-safety related control system failure
- safety/relief valve opening
- inadvertent RHR shutdown cooling operation
- generator load rejection with no turbine bypass combined with non-safety related control system failures
- turbine trip combined with non-safety related control system failures
- closure of main steam line isolation valves combined with non-safety related control system failures

- loss of condenser vacuum combined with non-safety related control system failures

- feedwater line break combined with non-safety related control system failures

- loss of instrument air combined with non-safety related control system failures

- large steam pipe break outside containment combined with non-safety related control systems failures

- loss of coolant accident inside containment combined with non-safety related control system failures, and

- main condenser offgas treatment system failure combined with non-safety related control system failures

All of the above cases were found to be bounded by the results of the relevant Chapter 15 analyses except for the loss of feedwater heating combined with turbine trip and no turbine bypass. The licensee submitted a GE analysis which shows that for a 100°F loss of feedwater heating combined with turbine trip and failure of the turbine bypass system no fuel cladding damage is predicted. The peak pressure is estimated at 1,250 psia which is below the ASME Code Section III Service Level B design limit of 1,375 psia. In addition, analysis reporting GE results show that for reactor operation at power levels lower than 95.6% of rated power, feedwater temperature reductions greater than 126°F will result in operation exceeding the MCPR safety limit, thus, can result into fuel damage (Ref. 7). Therefore, because the Clinton system design is such that a greater than 100°F feedwater temperature drop can occur, the licensee committed to implement (prior to the second cycle start-up) the following changes to decrease the likelihood of loss of feedwater heating and increase the indicating range of feedwater temperature inputs to the main control room:

- the licensee will institute operator procedures to shut the reactor down if feedwater heating delta-T approaches 100°F.

- the 48V DC and the AC power supplies will be coordinated to improve circuitry reliability

- the level trip setpoint for the extraction steam valves will be raised from 6.5 to 16.0 inches to allow level transients to be mitigated by automatic and operator actions prior to isolating the extraction steam flow to the heater drains

- during power ascension the control valves in the heater drain system will be "tuned" to ensure that their transient response is correctly adjusted, and

- the range of the feedwater temperature inputs to the main control room will be increased from a difference of about 115°F to about 250°F

We find the improvements in feedwater heating temperature monitoring, the improvements in the operation of extraction steam flow and the new operating procedures which instruct the operator to shut the reactor down if the feedwater temperature reduction is approaching 100°F to be acceptable.

3.0 INSTRUMENTATION AND CONTROL SYSTEMS EVALUATION

The "Combinatory Qualitative Event Analysis" postulated the possible failure modes of each nonsafety-related control system identified in Section 7.7 of the FSAR. The assumption was that all combinations of these nonsafety-related control system failures can occur to exacerbate the initiating event mechanisms identified in the FSAR Chapter 15, i.e., failure of common power bus, instrument sensor, or HELB. Each FSAR Chapter 15 event scenario was analyzed with this assumption to determine if the effects were beyond the bounds of the existing FSAR Chapter 15 analysis. The criteria for this determination were based on a "qualitative analysis" of how the nonsafety-related control system failures affect the reactor parameters. Those control systems whose failure would not affect reactor parameters were eliminated from further analysis. The following four control systems failures were found to affect reactor parameters, initiate engineered safety feature systems or trip nonsafety-related equipment.

1. Recirculation Flow Control
2. Feedwater Control
3. Pressure Regulator and Turbine-Generator Control
4. Anticipated Transient-Without-Scram (ATWS) Control

The staff requested the licensee to verify that all higher voltage power source failures were used in the analysis such that the loss of the higher voltage bus, as the common power source to various control systems, caused an event which was bounded by the existing analysis in Chapter 15 of the FSAR. The licensee was further requested to provide a positive statement regarding the requested analysis. The licensee's "Combinatory Qualitative Event Analyses" postulated failure of all nonsafety-related control systems regardless of the cause, i.e., failure or malfunction of its power sources or instrument power supplies. The effect of the failure of 120 volt AC to 6900 volt AC (including all intermediate AC voltages), and 125 volt DC, was included in the analysis. In addition to this analysis, a review was made to assure that no safety-related equipment, instrument or control systems were supplied from nonsafety-related AC or DC buses. Based on this qualitative analysis, the licensee has provided a positive statement as required by the staff. The statement assures that the failure of electric power, leading to multiple control system failures, would not result in an event which was not bounded by the FSAR Chapter 15 analysis.

In the conclusion section of their "Combinatory Qualitative Event Analysis" of the nonsafety-related control systems failure, IP provided the following statement.

> A further conclusion of this analysis is that multiple failures of nonsafety-related control systems at CPS do not impact the capability of safety-related systems, as required by NRC IE Notice 79-22. Furthermore, loss of electrical power to instrumentation and control systems does not affect the ability to achieve a cold shutdown condition, as required by NRC IE Bulletin 79-27.

The staff does not agree with this conclusion due to the following reasons.

1. IP's analysis of the nonsafety-related control system failure to determine if the consequences of these failures were within those analyzed in the FSAR Chapter 15 has no correlation with IE Bulletin 79-27 concerns.

   This Bulletin required licensees to review the effects of loss of power to each Class 1E and non-Class 1E bus supplying power to plant instrumentation and controls, and on the operator capability to achieve a safe (cold) shutdown condition using plant operating procedures following the power loss.

2. IP's analysis is combinatory qualitative which does not actually fail a bus to determine components, controls and instrumentation lost due to the bus failure. Rather all nonsafety-related control systems are failed regardless of the power supply. An analysis for IE Bulletin 79-27 concerns is a quantitative analysis where the affect of loss of each component, control and instrumentation supplied by a failed bus is evaluated.

   This discrepancy was discussed with the licensee in a telecon and it was agreed to consider the subject statement void.

4.0 CONCLUSION

We have reviewed the Illinois Power submittal providing information supporting deletion of license condition 4 for the Clinton Power Station. License condition 4 regards multiple non-safety control system failures, resulting from individual high energy line brakes. Analyses showed that the only case which is more severe than existing chapter 15 events is the loss of feedwater heating with turbine trip and no turbine bypass. Analyses further indicated that loss of feedwater heating up to 100°F with turbine trip and bypass failure is acceptable. The licensee committed to hardware and procedural changes which will minimize the probability for loss of feedwater heating and procedures instructing the operator to shut the reactor down in the event the feedwater heating temperature loss is approaching 100°F. Given that the loss of feedwater heating is a gradual and detectable change, operator action based on procedures is acceptable.

Based on the above evaluation, the staff concludes that the licensee's "Combinatory Qualitative Event Analysis" adequately addresses the staff's concerns regarding loss of electric power to the nonsafety-related control systems. The analysis has followed the guidelines provided in the staff's request for additional information and methodology approved in NUREG-0853, Supplement 6, and is, therefore acceptable.

5.0 REFERENCES

1. NUREG-0853, Supplement 6, "Safety Evaluation Report Related to the Operation of Clinton Power Station, Unit No. 1," July 1986

2. Letter from Gary M. Holahan, NRC to Frank A. Spangenberg, Illinois Power Company, dated April 17, 1987

3. Letter from D. L. Holtzscher, Illinois Power Company to USNRC "Clinton Power Station, License Condition 4, Control System Failures," dated November 18, 1988.

4. Letter form F. A. Spangenberg, Illinois Power Company to W. B. Butler, NRC, "Clinton Power Station Response to Request for Additional Information Related to Control Systems Failure," dated April 17, 1986.

5. Letter from F. A. Spangenberg, Illinois Power Company to W. R. Butler, NRC, "Clinton Power Station Control Systems Failure Analysis SER Outstanding Licensing Issue #15," dated May 15, 1986.

6. Letter from F. A. Spangenberg, Illinois Power Company to W. R. Butler, NRC, "Clinton Power Station, Control System Failure Reanalysis SER Outstanding Licensing Issue #15," dated July 15, 1986.

7. Letter from D. L. Holtzscher, Illinois Power Company to USNRC, "Clinton Power Station, License Condition 2.C(4), Control System Failure," dated March 20, 1989.

8. EAS-18-0388, "Special Transient Event Analysis to Support Control System Failure Analysis for Clinton Power Station" by S. Wolf and A. Horna, GE Nuclear Energy, dated March 1988.

9. LER 88-25, "Loss of Feedwater Heating System Transient Outside Design Basis Due to Inappropriate Level Controller Setting," by R. D. Freeman, dated July 28, 1988.

Dated: May 15, 1989