NRC Form 366 (9-83)

U.S. NUCLEAR REGULATORY COMMISSION
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LICENSEE EVENT REPORT (LER)

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
Cooper Nuclear Station	0 5 0 0 0 2 9 8	1 OF 0 5

TITLE (4): Error In Limiting Single Failure Assumption For The ECCS Performance Analysis

EVENT DATE (5)			LER NUMBER (6)			REPORT DATE (7)			OTHER FACILITIES INVOLVED (8)	
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	MONTH	DAY	YEAR	FACILITY NAMES	DOCKET NUMBER(S)
0 7	3 0	9 2	9 2	0 1 3	0 0	0 8	3 1	9 2		0 5 0 0 0
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OPERATING MODE (9): N

POWER LEVEL (10): 1 0 0

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more of the following) (11)

[ ] 20.402(b)	[ ] 20.405(c)	[ ] 50.73(a)(2)(iv)	[ ] 73.71(b)
[ ] 20.405(a)(1)(i)	[ ] 50.36(c)(1)	[X] 50.73(a)(2)(v)	[ ] 73.71(c)
[ ] 20.405(a)(1)(ii)	[ ] 50.36(c)(2)	[ ] 50.73(a)(2)(vii)	[ ] OTHER (Specify in Abstract below and in Text, NRC Form 366A)
[ ] 20.405(a)(1)(iii)	[X] 50.73(a)(2)(i)	[ ] 50.73(a)(2)(viii)(A)	
[ ] 20.405(a)(1)(iv)	[X] 50.73(a)(2)(ii)	[ ] 50.73(a)(2)(viii)(B)	
[ ] 20.405(a)(1)(v)	[ ] 50.73(a)(2)(iii)	[ ] 50.73(a)(2)(x)	

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER
John R. Myers	AREA CODE 4 0 2 8 2 5 - 3 8 1 1

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT (13)

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS	CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS

SUPPLEMENTAL REPORT EXPECTED (14)

[ ] YES (If yes, complete EXPECTED SUBMISSION DATE) [X] NO

EXPECTED SUBMISSION DATE (15): MONTH | DAY | YEAR

ABSTRACT (Limit to 1400 spaces, i.e., approximately fifteen single-space typewritten lines) (16)

On July 30, 1992, a nonconservative assumption in the ECCS Performance/LOCA Analysis was discovered. This analysis was originally conducted to demonstrate compliance with 10 CFR 50.46 under postulated design basis Loss of Coolant Accident conditions. Accordingly, at 5:30 p.m., a plant shutdown was initiated in accordance with Technical Specification 1.0.J and a Notification of Unusual Event was declared. The nonconservative assumption was that the most limiting single failure was the loss of one Low Pressure Coolant Injection (LPCI) Subsystem injection valve. A review, conducted as a part of the Design Basis Reconstitution Program for Cooper Nuclear Station, revealed that several single failure modes exist for the 125 VDC Power System which could result in the loss of one Emergency Diesel Generator, two of the four LPCI pumps, a Core Spray pump , and one Reactor Recirculation System discharge valve, in addition to the loss of one LPCI injection valve. A General Electric Company analysis of ECCS performance with the new limiting single failure resulted in Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) restrictions for continued power operation. Provided the MAPLHGR limits are observed, the GE analysis demonstrated that the requirements of 10 CFR 50.46 would be met. At approximately 8:00 p.m. on July 30, 1992, the plant shutdown was discontinued after reaching about 55 percent power.

The single failure analysis performed in 1976 by the Architect/Engineer contained an erroneous evaluation of the ECCS equipment affected by a 125 VDC Power failure. This resulted in the invalid, nonconservative single failure assumption used in the ECCS Performance/LOCA Analysis. Restricted power operation has continued while complying with the new MAPLHGR limits. Plant modifications will be made to restore the validity of the previous assumptions for the ECCS Performance/LOCA Analysis.

A. Event Description

On July 30, 1992, at 5:30 p.m., a plant shutdown was initiated in accordance with Technical Specification 1.0.J (equivalent to Standard Technical Specification 3.0.3), and a Notification of Unusual Event was declared. This resulted from the discovery of a nonconservative assumption in the Emergency Core Cooling System (ECCS) Performance Analysis, under postulated design basis Loss of Coolant Accident (LOCA) conditions, conducted to demonstrate compliance with 10 CFR 50.46. The nonconservative assumption was that the most limiting single failure was the failure of one Low Pressure Coolant Injection (LPCI) Subsystem injection valve. A review, conducted as a part of the Design Basis Reconstitution Program, revealed that several failure modes exist for the 125 VDC Power System which would result in a more limiting single failure.

The failure of one division of the 125 VDC Power System could result in the loss of one Emergency Diesel Generator, two of the four LPCI pumps, a Core Spray pump, and a Reactor Recirculation System discharge valve, in addition to the loss of one LPCI injection valve. This discovery placed the plant in a condition outside of the design basis of the plant.

The General Electric (GE) Company analyzed ECCS performance with the new limiting single failure, loss of one division of 125 VDC Power. This analysis resulted in Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) restrictions for continued power operation. Provided the MAPLHGR limits are observed, the GE analysis demonstrates that the requirements of 10 CFR 50.46 are met and the plant is within its design basis. At approximately 8:00 p.m. on July 30, 1992, the plant shutdown was discontinued at about 55 percent power.

B. Plant Status

The plant was in normal full power operation at the time this condition was discovered. The condition has existed since the LPCI Loop Selection Logic modification was completed in 1976.

C. Basis for Report

Discovery of a nonconservative assumption in the 10 CFR 50.46 ECCS Performance/LOCA Analysis, resulting in the plant being in a condition outside the design basis of the plant. This condition is also reportable as a condition which alone could have prevented the fulfillment of the safety function of systems required to remove residual heat, and is also considered to violate Technical Specification 3.5.F.3. This resulted in a Technical Specification initiated shutdown which was terminated prior to completion and declaration of a Notification of Unusual Event. Therefore, this report is being submitted in accordance with 10 CFR 50.73 (a)(2)(i), (a)(2)(ii), and (a)(2)(v).

D. Cause

Cooper Nuclear Station is required, as part of its licensing basis, to be able to mitigate the effects of a number of postulated accidents, including a LOCA. 10 CFR 50.46 provides the current acceptance criteria for evaluating the performance of the Emergency Core Cooling Systems under postulated design basis LOCA conditions. 10 CFR 50 Appendix K describes the requirements placed on evaluation models used for analyzing accident scenarios, to ensure that the criteria given in 10 CFR 50.46 are met under worst case LOCA conditions. In addition, 10 CFR 50 Appendix K requires that the analysis assume that the most limiting single failure of ECCS equipment has taken place. The CNS licensing basis also assumes that a loss of offsite power occurs coincident with the LOCA.

When CNS was originally licensed, the acceptance criteria was much less restrictive. The original licensing basis required sufficient cooling of the reactor core under postulated LOCA conditions with diversity and redundancy of cooling mechanisms such that the core geometry remained defined. To meet this criteria, a Peak Cladding Temperature (PCT) of 2700 degrees Fahrenheit was established in the late 1960's, and it was demonstrated for the design basis LOCA (the rapid circumferential break of the reactor recirculation pipe) that this criteria was satisfied by diverse phenomena, spraying or flooding. In order to satisfy the diversity criteria, the Core Spray and Low Pressure Coolant Injection (LPCI) systems were evaluated independently of each other. For the LPCI System to inject sufficient water to flood the core, flow from three of the four pumps was required. To accomplish this, flow from all LPCI pumps was directed to the unbroken Reactor Recirculation System loop by the LPCI loop selection logic. By the very nature of this design, a single component failure could negate the flooding mechanism. However, independent protection was provided by the Core Spray System.

With the advent of the Interim Acceptance Criteria, and now, the current ECCS acceptance criteria (10 CFR 50.46), a more restrictive PCT limit was established, i.e., PCT less than 2200 degrees Fahrenheit. In addition, the new limit had to be met assuming the most limiting single failure, which, based on the loop selection design, was the failure of a LPCI injection valve, with a loss of offsite power. In addition, the ECCS evaluation model was required to be consistent with the requirements of 10 CFR 50 Appendix K, which describes acceptable evaluation methodology. The net effect of the new criteria and requirements was to reduce the plant operating flexibility, and in some BWR/4 cases, introduce power restrictions. To offset these effects, a modification to the ECCS was developed to improve ECCS performance.

D. Cause (Continued)

In response to the new rule, Cooper Nuclear Station was required to perform a plant modification, commonly called the LPCI modification. The major changes included cross-powering two of the LPCI pumps and converting the LPCI injection valves (RHR-MO-MO25A, B) and the Reactor Recirculation System discharge valves (RR-MO-MO53A, B) to 250 VDC power. At the same time, control logic for the LPCI injection valves and the Reactor Recirculation discharge valves was powered from the appropriate division of the 125 VDC System. This increased the reliability and performance of the ECCS systems in responding to design basis accidents.

As a part of the modification and in response to the new ECCS performance rules, a Single Failure Analysis was also conducted to identify the limiting component failures for both the Reactor Recirculation System suction and discharge line breaks. The Single Failure Analysis was performed by the Architect/Engineer for Cooper Nuclear Station in 1976. The results of this analysis showed that the failure of the LPCI injection valve continued to provide the minimum available ECCS equipment for mitigating the consequences of the design basis LOCA. This information was then provided to the General Electric Company. GE used this single failure analysis input to evaluate compliance with 10 CFR 50.46 under design basis LOCA, loss of offsite power and limiting single failure conditions.

In reviewing the configuration of the control power for the station as part of Design Basis Reconstitution, it was discovered that the failure of 125 VDC control power coincident with specific Reactor Recirculation System suction and discharge line breaks would result in fewer ECCS pumps injecting coolant into the core than was previously assumed. Specifically, it could be postulated that for a Reactor Recirculation System discharge line break and loss of 125 VDC control power on the opposite side (e.g., Reactor Recirculation System loop A break, Division II control power failure), only one Core Spray pump would be available to provide cooling water to the vessel. The GE ECCS Performance/LOCA Analysis previously assumed and analyzed configuration was one Core Spray pump and one LPCI pump, or two Core Spray pumps. This indicated an error in the previous Single Failure Analysis performed by the Architect/Engineer.

A similar concern was identified for a Reactor Recirculation System suction side break, wherein only one Core Spray and one LPCI pump would be available. The GE ECCS Performance/LOCA Analysis previously assumed and analyzed a minimum of one Core Spray and two LPCI pumps available. The conditions discussed above are independent of offsite power availability.

The single failure analysis performed in 1976 by the Architect/Engineer contained an erroneous evaluation of the ECCS equipment affected by a 125 VDC Power failure. This resulted in the invalid, nonconservative single failure assumption used in the ECCS Performance/LOCA Analysis.

E. Safety Significance

The safety significance of the nonconservative assumption in the ECCS Performance/LOCA Analysis is that with the operating limits in effect prior to the GE re-analysis, the calculated peak cladding temperature could have exceeded 2200 degrees Fahrenheit, given a postulated design basis LOCA and loss of one division of 125 VDC power. This analytical nonconservatism only has safety significance in the highly unlikely event that a Reactor Recirculation System line break occurs simultaneously with a failure in the division of 125 VDC power that controls ECCS equipment in the loop opposite the Recirculation Loop that failed.

The actual safety significance is considered to be minimal. The likelihood of the events described above occurring simultaneously is very small (estimated by the PRA Group to be on the order of 1 X E-8). In addition, conservatism is built into the computer models, peak cladding temperature limits, and input parameters to the computer model. This conservatism serves to illustrate that the actual safety significance of the invalid single failure assumption is minimal.

F. Safety Implications

The plant response to the large break LOCA is most severe at 100 percent power, as the plant was when this condition was discovered. As such, there are no safety implications beyond those discussed in Paragraph E above.

G. Corrective Action

Upon determining that this condition existed, a power reduction was commenced, toward achieving hot shutdown within 6 hours and cold shutdown within 30 hours, as required by CNS Technical Specifications, and a Notification of Unusual Event was declared. Prior to achieving hot shutdown, at approximately 55 percent power, the results of a General Electric analysis indicated that the 10 CFR 50.46 requirements and design basis for the ECCS Systems would be met with certain operating restrictions. The present operating restrictions of 90 percent power will remain in effect until modifications are completed to restore the validity of the original assumptions used in the ECCS Performance/LOCA Analysis.

Modifications will be made that will reduce the impact of the failure of one division of 125 VDC Power on the ECCS equipment. This will be accomplished by modifying the control power for the LPCI System injection valves and the Reactor Recirculation discharge valves. The failure of the 125 VDC control power will be less severe after this modification, because the LPCI injection valves and Reactor Recirculation discharge valves will not be affected. After the modification, the ECCS equipment available to cope with the consequences of a design basis LOCA and the concurrent loss of offsite power will be bounded by the existing ECCS Performance/LOCA Analysis, whether the single failure is the LPCI injection valve, the loss of one 125 VDC source, or the loss of one 250 VDC source.

H. Similar Events

None.