U.S. NUCLEAR REGULATORY COMMISSION

REGION III

Report No. 50-373/88008; 50-374/88008

Docket No. 50-373; 50-374 License No. NPF-11; NPF-18

Licensee: Commonwealth Edison Company
P. O. Box 767
Chicago, IL 60690

Facility Name: LaSalle County Station, Units 1 and 2

Inspection At: LaSalle Site, Marseilles, IL

Inspection Conducted: March 16 through 24, 1988

Inspectors: NRC Augmented Inspection Team

Team Leader: M. A. Ring M. A. Ring 5/11/88
Date

Team Members: R. A. Kopriva M. A. Ring for 5/6/88
Date

L. E. Phillips M. A. Ring for 5/11/88
Date

P. Shemanski M. A. Ring for 5/11/88
Date

B. A. Azab M. A. Ring for 5/6/88
Date

Approved By: W. L. Forney, Chief
Reactor Projects Branch 1
Date

Inspection Summary

Inspection on March 16 through 24, 1988 (Report No. 50-373/88008(DRP); 50-374/88008(DRP))
Areas Inspected: Special Augmented Inspection Team (AIT) inspection conducted in response to the dual recirculation pump trip and subsequent core power

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oscillations resulting in a reactor trip on March 9, 1988, at LaSalle, Unit 2. The review included root cause determination, safety significance, performance of operators and equipment, adequacy of procedures, effects on the reactor, reporting actions and potential generic implications.

Results: No violations or deviations were identified; however, the licensee has committed to procedure and Technical Specification changes as well as further study in the areas of inherent shutdown mechanisms, instrumentation capability and uncertainties in the decay ratio calculations. The licensee's interim report, as required by the CAL, is included as attachment 5 to this report.

Augmented Inspection Team Report

Attachments

Attachment No.	Description
1	Confirmatory Action Letter (CAL-RIII-88-03)
2	Augmented Inspection Team (AIT) Charter
3	Figures
Figure 1	BWR Power to Flow Map
Figure 2	Decay Ratio
Figure 3	Startrec Traces - Beginning of Event
Figure 4	Startrec Traces - Oscillations
4.	Technical Specifications
5	Commonwealth Edison Company (CECo) Response to CAL item 5, dated April 15, 1988
6	LER 88-003-00
7.	GE SIL 380

I. INTRODUCTION

A. Synopsis of Event

On Wednesday, March 9, 1988, around 5:30 p.m. CST, the LaSalle Unit 2 reactor was operating at steady state conditions at approximately 84% power. Instrument Maintenance Department personnel were in the process of performing an instrument surveillance when a valving error produced a pressure pulse which actuated the instrumentation which causes a trip of both recirculation (RR) pumps in order to decrease power in the event of an Anticipated Transient Without Scram (ATWS). Both RR pumps tripped causing a flow and power decrease. Control rods remained in the high power (99% Flow Control Line (FCL)) position. As a result of the rapid power decrease caused by the trip of the RR pumps, the feedwater heater level control system was unable to control level in the feedwater heaters and began isolating extraction steam from the heaters. This resulted in cooler feedwater being supplied to the reactor. Approximately five minutes after the RR pump trip, operators observed the Average Power Range Monitor (APRM) indication in the control room to be oscillating between 25% and 50% power every 2 to 3 seconds. Approximately seven minutes after the RR pump trip, as operators were attempting to restore forced flow and making preparations to scram, the reactor automatically scrammed on high neutron flux as seen by the APRMs. At 6:32 p.m. CST, the licensee notified the NRC of the RR pump trip, the loss of feedwater heating, and the resultant scram.

B. AIT Formation

At the time of the event on March 9, 1988, the Resident Inspector assigned to LaSalle was offsite attending the Resident Seminar and the Senior Resident Inspector position for LaSalle was vacant due to a recent promotion. The initial licensee report on the event did not discuss the flux oscillations but indicated that the event was still being investigated. Upon further investigation and appreciation of the magnitude of the oscillations a Region III morning report update of the event was issued on March 15, 1988. On March 16, 1988, an Augmented Inspection Team (AIT) was formed which included three Region III individuals; M. A. Ring, Chief, Reactor Projects Section 1B and Team Leader, R. A. Kopriva, LaSalle Resident Inspector, and B. A. Azab, Reactor Safety Inspector, and two NRR individuals; L. E. Phillips, Senior Nuclear Engineer, and P. Shemanski, LaSalle Project Manager. All of the AIT members had arrived onsite by the morning of March 17, 1988. Concurrent with the AIT activities, Region III issued a Confirmatory Action Letter (CAL-RIII-88-03) which was received by the licensee on March 17, 1988, and is included as Attachment 1 to this report. The CAL confirmed certain actions to be taken by the licensee in support of the AIT and established conditions to be met prior to the restart of LaSalle, Unit 2.

C. AIT Charter

On March 17, 1988, a draft charter for the AIT was formulated and transmitted to the AIT onsite (Attachment 2 to this report). The general areas to be investigated were:

- Sequence of events
- Core performance during the event
- Operator performance
- Procedure adequacy
- Reactor effects
- Reporting

D. Persons Contacted

Commonwealth Edison Company

*G. J. Diederich, Station Manager
*N. Kalivianakis, General Manager, BWR Operations
*D. Galle, Vice President, BWR Operations
*H. E. Bliss, Manager of Nuclear Licensing
*W. R. Huntington, Services Superintendent
*T. Rausch, Nuclear Fuel Services
*W. F. Naughton, Nuclear Fuel Services Manager
*M. Turbak, Assistant Licensing Manager
*J. Bitel, Manager Nuclear Safety
*R. J. Raguse, Production Training Supervisor
*T. Shaffer, Training Supervisor
*R. O. Armitage, Lead License Instructor
*K. W. Peterman, Nuclear Fuel Services
*L. H. Lauterbach, Onsite Nuclear Safety Supervisor
*H. McLain, Onsite Nuclear Safety
*W. S. Marcus, Engineering-Site Supervisor
*J. C. Renwick, Production Superintendent
*J. A. Miller, Technical Staff
*M. H. Richter, Assistant Technical Staff Supervisor
*D. A. Brown, Quality Assurance Superintendent
*P. F. Manning, Assistant Superintendent - Technical Services
*T. A. Hammerich, Technical Staff Supervisor
*A. C. Settles, Regulatory Assurance
B. S. Westphal, Operating Engineer
R. W. Stobert, Director of Quality Assurance Operations
J. A. Silady, Nuclear Licensing
M. Wagner, Dresden Nuclear Group
M. G. Santic, Master Instrument Engineer
L. W. Raney, Nuclear Safety Braidwood
R. Weidner, Production Training
J. Dedin, Production Training
R. Graham, Nuclear Station Operator
E. McVey, Technical Staff

General Electric Company (GE)

H. Pfefferlen, Licensing
G. A. Watford, Engineering

*Denotes those attending the exit meeting on March 24, 1988.

In addition, several other members of the LaSalle staff were contacted by the AIT.

II. DESCRIPTION - DUAL RECIRCULATION PUMP TRIP EVENT OF MARCH 9, 1988

A. Narrative Description

On Wednesday, March 9, 1988, around 5:32 p.m. CST, the LaSalle Unit 2 reactor was operating at steady state conditions at approximately 84% power with 76% rated core flow using both recirculation (RR) pumps and with the control rods withdrawn to the 99% flow control line. Feedwater temperature was 402°F. LaSalle Unit 1 was operating at power in steady state conditions and was unaffected by the subsequent events on Unit 2. Instrument Maintenance Technicians (IMs) were in the process of performing a surveillance test on Wide Range level instrument 2B21-N037BB to check the Reactor Core Isolation Cooling (RCIC) initiation function at -50 inches reactor level. The IMs were stationed at the instrument rack and in the control room and had received permission from the appropriate operations personnel to perform the surveillance. The IM at the instrument rack had correctly isolated and equalized the instrument (2B21-N037BB) in accordance with the functional test procedure, LIS-NB-404. The next action was to open the test/vent valves, however, instead the IM technician opened the isolation valves to the variable and reference legs to the instrument. Since the equalizing valve was still open, a pressure equalization occurred between the variable and reference legs for this instrument and all the other instruments which share the same reference leg. At the time of the valving error feedwater level control was selected to channel B which takes input from an instrument which utilizes the same reference leg as 2B21-N037BB. The equalization produced by the valving error resulted in a high "indicated" level to feedwater level control, causing the operating feedwater pumps (A turbine driven reactor feedwater pump - TDRFP, and the motor driven reactor feedwater pump - MDRFP) to begin reducing flow. The IMs realized a valving error had been made and attempted to correct the error by shutting the reference and variable leg isolation valves. This action caused a pressure pulse on the reference leg of all the instruments which share the same reference leg at that instrument rack. Increasing pressure on the reference leg caused the level instruments to indicate low reactor vessel level. The key instruments which were affected by this pulse were the ATWS RR pump trip switches 2B21-N036C and 2B21-N036D, which are designed to trip the A and B RR pumps to off. Both RR pumps did, in fact, trip off. Instrument 2B21-N024B which provides a reactor protection system

(RPS) channel B1 low level 1/2 scram was also affected and resulted in a 1/2 scram signal and the associated alarm.

The trip of the RR pumps resulted in a large and rapid power reduction (approximately 45%) as a result of the large flow reduction (to natural circulation conditions). The control rods remained in their pre-RR pump trip position on the 99% control line (see Attachment No. 3-BWR Power to Flow Map). This region of the BWR Power to Flow Map was known to be susceptible to instabilities in some BWRs. As a result of the large drop in power, a large drop in steam flow occurred causing large changes in extraction steam flow and extraction steam pressure. Extraction steam supplies the heating to the feedwater heaters. The changes in extraction steam caused severe perturbations in the feedwater heater level control system due to water flashing to steam from lower shell pressures, reductions in shell side input from reduced steam flow, and changes in condensing rate. The feedwater heater level control was unable to react fast enough to control the large load reduction and tripped the extraction steam input to the heaters in order to prevent induction of water into the main turbine. The securing of steam heating to the feedwater heaters resulted in cooler feedwater being supplied to the reactor (approximately 45°F decrease in 4 minutes) which is the equivalent of a positive reactivity addition. This resulted in an increased power to flow ratio which further reduced the margin to instability.

At this point in the event, the operators in the control room were primarily concerned with attempting to restore feedwater heaters. The operators had correctly determined that an ATWS event had not occurred but that an instrument problem had resulted in the loss of both RR pumps. The loss of feedwater heating was not unexpected for the large power drop caused by the RR pump trip. The operators also realized that the reactor was operating in a region of the power to flow map where instability was possible. Between 4 and 5 minutes into the event, the Average Power Range Monitor (APRM) indications were observed by the operators to be oscillating between 25% power and 50% power every 2 to 3 seconds and the Local Power Range Monitor (LPRM) down scale alarms began to annunicate and clear. (Later examination of the STARTREC, Startup Transient Recorder, (a high speed, multi-channel recording system installed for startup testing which starts recording when selected parameters exceed predetermined limits) showed the oscillations to be much larger than the operators were able to see). The APRM indications confirmed the onset of instabilities and the operators attempted to restart a RR pump in order to increase flow to leave the instability region. Attempts to start a RR pump were unsuccessful and the shift commenced preparations to manually scram the reactor. About 7 minutes into the event and before the shift was able to perform a manual scram, the reactor automatically scrammed on high neutron flux as seen by the APRMs. The scram shutdown the reactor as designed and recovery from the scram proceeded normally. Some minor equipment problems occurred during recovery and subsequent [illegible] shutdown, however, these were judged by the AIT to have no effect on the event and will not be discussed further in this report. The licensee informed the NRC at 6:32 p.m. CST of the RR pump loss, feedwater heating loss and resultant scram.

B. Sequence of Events

Times and sequences of events in the previous narrative description were derived by the AIT from a combination of interviews and hard data. The following sequence of events represents a compilation of information by the AIT taken from the alarm printer, the Startrec recording system and interviews with licensee personnel. Times are given in 24 hour clock time (17:32 equals 5:32 p.m.) and are all Central Standard Time.

SEQUENCE OF EVENTS FOR MARCH 9, 1988
LASALLE UNIT 2 INSTABILITY EVENT

Initial Conditions

- 84% Reactor Power (930 MWe)
- Steady State Conditions
- 99% Flow Control Line
- 76% Rated Core Flow (82×10^6 lb/hr)
- Feedwater Temperature = 402°F
- LIS-NB-404 in process (Surveillance that tests RCIC initiation at -50" reactor water level.)

Event Summary

March 9, 1988

Time	Event
17:32 (approximately)	Instrument Maintenance Technician valved in the variable and reference legs of a differential pressure switch with the equalizing valve open; initiating a pressure equalization between the two legs and a high "indicated" reactor level.
17:32:33	High Reactor Water Level Alarm initiated.
	STARTREC (Startup Transient Recorder) initiated on increase in narrow range level and ran for programmed 1 minute duration.
*	Instrument Maintenance Technician corrected valving error by isolating reference leg from variable leg which resulted in a low "indicated" level spike causing other instrumentation to actuate.

17:32:49	2A/2B ATWS alarm initiated a trip of both Reactor Recirculation (RR) pumps and power and flow began coasting down to natural circulation levels. Division 2 Reactor Low Level Alarm initiated.
	2A ATWS cleared.
17:32:50	Half scram on +12.5" reactor water level initiated.
	2B ATWS cleared.
	Division 2 Reactor Lo Level Alarm cleared.
	Reactor Water Level 3 Alarm (+12.5") initiated.
17:32:51	Reactor Water Level 3 Alarm cleared.
	Reactor Water Level Hi Channel B trip.
17:32:52	Reactor Water Level Hi Channel B was manually reset.
*	Half Primary Containment Isolation System (PCIS) level trip was manually reset.
	Nuclear Station Operator (NSO) saw that B narrow range reactor water level indicator was approximately 30" and rising while A and C were steady at approximately 40".
17:33:10	First feedwater heater high level alarm annunciated.
17:33:20	First feedwater heater isolates. Feedwater heaters continue to isolate for duration of event.
*	Unit 2 NSO reviewed feedwater heater situation and planned to reopen extraction steam valves after valves fully closed to regain feedwater heating.
	Shift foreman discharged to local heater controllers to aid in reestablishing feedwater heating.
17:36(approximately)	Shift engineer entered control room.
	Operators observed APRMS oscillating between 25% and 50% power with an approximate 2-3 second period.

*	Operators attempted to restart the RR pumps per abnormal operating procedure, LOA-RR-07.
17:36:55	B RR flow control valve locked up.
*	Equipment Operator discharged to reset the lockouts on flow control valves.
17:37:21	First LPRM downscale alarm annunciated. The LPRM downscale alarms continued to flash and clear, on a 2 second period.
17:37:51	First LPRM Hi Alarm annunciated.
17:38:20	The A flow control valve was reset.
	STARTREC initiated on increase in narrow range level and ran for 1 minute.
*	NSO ramped the A flow control valve to minimum position.
17:39(approximately)	NSO attempted restart of 2A RR pump twice, but was unsuccessful.
*	Shift Engineer directed a manual scram to be initiated.
17:39:19	STARTREC terminated its second 1 minute run.
17:39:23	Channel A neutron APRM trip.
17:39:25	Channel B neutron APRM trip.
	Reactor scrammed on 118% neutron flux.

NOTE: * Designates that no Hathaway Recorder time was available for the event. However, the event is listed in the approximate sequence in which it occurred.

III. INVESTIGATIVE EFFORTS

A. Synopsis of AIT Activities

The AIT members had all arrived onsite on March 17, 1988, and were thoroughly briefed on the event by the licensee and General Electric personnel in a meeting at the site at 1:00 p.m. hours on March 17. The team was provided with pertinent instrument records of the event (including Sequence of Events data), and with documentation comprising the safety evaluation by the licensee. The latter included correspondence between the licensee and the reactor vendor (GE). In response to a team request, the licensee also provided a written description of the operator response and assessment of the event as it occurred.

The inspection team had various meetings with the LaSalle plant management and staff and with GE supporting staff during the assessment of the event. Subsequent to telephone conferences with Headquarters and Region III offices, the plant was allowed to restart at approximately 10:45 p.m. on March 17. Operation was to be under operating procedures which had been modified by a standing order to require manual scram after trip of the recirculation pumps in Operating Modes 1 or 2.

The AIT investigation continued with interviews of control room operating personnel and a walkdown of the control room response to the event. The AIT documented several issues and concerns which were presented to the licensee at a preliminary exit meeting on March 18. The AIT concluded onsite activities with an exit meeting on March 24, 1988.

B. Core Nuclear and Thermal Hydraulic Performance

1. Core Performance

In general, the AIT confirmed the adequacy of the assessment of core performance performed by the licensee and the reactor vendor. Several concerns and questions, however, were developed by the AIT and these are discussed in the Concerns and Recommendations portion of this report (paragraph IV). The following paragraphs provide a discussion of core performance during the event.

Following the trip of the RR pumps, Core Thermal Power (CTP) decreased and stabilized within about 30 seconds at about 40%. The APRMs showed stable indications (the APRMS read neutron flux as distinguished from CTP, however, both were stable at this point). As feedwater temperature decreased, CTP increased slightly to 43%. At approximately 4.8 minutes after the RR trip, the APRMs began oscillating and the LPRM down scale alarms were received. At 5.8 minutes after the RR pump trip, the STARTREC system initiated its second recording for the designed 1 minute period and stopped about 8 seconds before the full scram.

STARTREC information is not available to the operators in the control room at the time of recording, so from the APRM recorders in the control room, the operators believed the oscillations were approximately 25% power (neutron flux) in magnitude (between 25% and 50% power) every 2 to 3 seconds. Analysis of the STARTREC traces showed APRM peak to peak oscillations ranging from 20% to about 95% power. Extrapolation of the traces to the time of the scram leads the AIT to believe the oscillations were at least 100% peak to peak when the scram occurred. The oscillation frequency was approximately 0.45 hz. The approximately 2 second period of the oscillation is driven by core flow hydraulic conditions, primarily the time it takes a void perturbation to travel the length of the fuel . The APRMs measure neutron flux and during reactivity changes the neutron flux leads the fuel cladding heat flux by the thermal time constant of the fuel pellet/pellet-clad-gap arrangement. This time constant is approximately 6-7 seconds and acts to filter neutron flux spikes. Consequently, the heat flux oscillations for this event are estimated to be less than 10% of the neutron flux oscillations cycling around an average CTP of about 45% during the final minute of STARTREC recording. Based on LPRM alarm signals at 87% APRM power and LPRM readings after pump trip, the AIT estimates that the peaking factor increased from 2.11 prior to the event to a value of 2.65 at the time of the APRM alarm. This corresponds to a peak neutron flux level of 312% (relative to rated core average) at the 118% APRM scram level. However, because of the smaller changes in heat flux, the 13.4 Kw/ft fuel design limit was not exceeded and no core damage was indicated by chemistry results.

One of the more important considerations in this type of event is whether the LPRM swings are in phase with the APRMs or not. The reason for this concern is that the core protection actions are actuated by the APRM signals, so, if some LPRMs were oscillating out of phase with the core average, the effect would be to lower the APRM signals that these LPRMs feed. Consequently, the effectiveness of the APRMs as a protective system would be less and local areas of the core would be undergoing oscillations of much greater magnitude than indicated by the APRMs. Analysis of the alarm printout of the LPRM downscale and LPRM hi alarms, the "clean" sinusoidal wave shape, and the in phase APRM traces from STARTREC by the licensee and the reactor vendor (GE) determined that the LPRMs and APRMs were in phase with each other. The AIT verified this analysis and concurred with the licensee's interpretation. This type of oscillation is less severe with lower power peaks to trip than would occur with regional oscillations which have been observed in foreign reactors. Generic analyses performed during the resolution of Generic Issue B-19 bound the LaSalle Unit 2 instability and demonstrated that the fuel thermal or mechanical limits were not exceeded during the event.

General Electric's evaluation of the LaSalle event concluded that the frequency and magnitude of oscillations which Unit 2 experienced were consistent with the characteristics observed during stability testing and operation at other BWRs. GE further concluded that the event was bounded by the generic analysis and that no fuel thermal or mechanical limits were exceeded during the event. While the AIT did not agree or disagree with the GE conclusions at the time of exit, there were several questions and concerns relating to unexpected aspects of the event which required further study. These included the following:

(1) failure to predict the susceptibility of the Unit 2 core to thermal hydraulic oscillations based on the calculated decay ratio.

(2) magnitude of the oscillations compared to previous events with similar behavior,

(3) potential for out of phase regional oscillations of much larger magnitude and the lack of a defined boundary based on inherent shutdown mechanisms,

(4) questions about the adequacy of instrumentation for detection, suppression, and evaluation of limit cycle neutron flux oscillations, and

(5) questions about the adequacy of technical specifications and procedures for detection and suppression of neutron flux oscillations.

These issues are discussed in more detail in paragraph IV, "Concerns and Recommendations".

2. Chemistry Results

Following the scram on March 9, 1988, the LaSalle Station Chemistry Department took post shutdown samples of the reactor coolant water in order to determine if there were any indications of fuel damage. Analysis of the iodine results from this sample indicated no abnormalities (between 2×10^{-3} and 1×10^{-5} microcuries per gram for iodine 131 through 135). The reactor water dose equivalent I-131 was less than 2×10^{-4} microcuries per gram as compared to a Technical Specification limit of 0.2 microcuries per gram. This data, as well as the past two months sample data for both Units 1 and 2, was made available to the AIT for verification by the licensee. As a result of the CAL, the licensee developed an increased frequency sampling program of reactor water and off gas which was implemented following restart of Unit 2. The results of this sampling are documented in Attachment 5 and show no indications of fuel damage or abnormalities from readings prior to the event.

3. Equipment Performance

The following paragraphs summarize the AIT's conclusions regarding the performance of specific systems or pieces of equipment during the event. By the CAL, the licensee was also requested to address equipment performance and that assessment is included in Attachment 5.

(a) Recirculation Pumps and Flow Control Valves

A trip of both RR pumps is a designed feature of the LaSalle plant in order to cause a power reduction in the event of an ATWS, as indicated by a loss of reactor level without an associated scram. The pressure pulse on the reference leg of the ATWS switches appeared to be sufficient to provide this indication and consequently the trip of both pumps occurred as would be expected.

During the RR pump trip recovery, the operators attempted to restart the RR pumps and were unsuccessful. At the time of the exit, the RR pump start failure was believed to be due to the failure to satisfy one of the pump start interlocks. Exactly which interlock was not satisfied was not conclusively determined, however, likely candidates include the recirculation flow control valve not fully in minimum position and the hi speed start permissive (depends on feedwater flow). The operators have no indication available in the control room to determine which RR pump start interl
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positic... ... FCV was not reset because there was an abnormal signal alarm which would have required additional operator actions. FCV lockout was not fully investigated by the AIT.

MARK FCV Lock Out or Opinion

b. Feedwater Heaters

Shortly after the RR pump trip, the feedwater heater level control system began isolating extraction steam from the heaters in order to protect the turbine from water induction. The resultant cooler feedwater which was supplied to the reactor caused the reactor to become even more susceptible to instability. The AIT reviewed the feedwater and feedwater heater performance and concluded that the feedwater heater actions were as expected and had been seen before for similar large rapid drops in load. A more complete description is provided in Attachment 5. This observation, however, negated an initial licensee statement that two abnormal conditions, operation at high power in natural circulation and reduced feedwater heating, were necessary to put the reactor into the unstable region, in that, feedwater heating would be expected to be lost every time both RR pumps were lost from high power. The AIT believed the licensee efforts in investigating the performance of the feedwater system were appropriate.

c. Power to Flow Scram (APRM flow-biased trip)

Since the reactor ends up at lower power following a RR pump trip, and the reactor eventually scrammed on high power in terms of APRM flux, concern was expressed over the operation of the power to flow or APRM flow-biased trip.

The APRM flow-biased trip is designed to protect against spurious scrams due to transient events that cause spikes in neutron flux. APRM neutron flux correlates to thermal power level during steady state conditions. However, during power increase events the APRMs will over-predict thermal power because the neutron flux leads the reactor heat flux due to the thermal time constant of the fuel and cladding. It should be noted that no credit is taken for the flow-biased scram in the transient analysis; only the 120% high flux scram is credited.

The APRM flow-biased trip circuitry receives the APRM neutron flux signal and filters it through an R-C circuit with a 6 second time constant. This R-C circuit which is known as the "thermal power monitor" essentially integrates the APRM neutron flux signal over the past 6 seconds and develops an APRM reading which simulates thermal power by accounting for the lag in response of the fuel cladding heat flux to neutron flux variations. The output of the thermal power monitor is then compared to the flow-biased scram setpoint which is calculated by the equation: $S=0.66\ W_D + 51\%$. W_D is the driving flow which that at 6.9 minutes following the RR pump trip, RPS APRM channels A and B actuated to produce the full scram. APRMs A, C, and E all indicated neutron monitoring trip and are grouped as channel A in the logic scheme. No trip indications appeared for the D and F APRMs which is believed to be due to the fact that the B APRM signal completed the logic and actuated the scram. The licensee's review included a functional check of the APRM trip setpoints following the event, a comparison utilizing gain adjustment factors, and an evaluation of whether the flux oscillations could have been missed by the scram logic. The AIT concurred in the licensee's conclusion that the Reactor Protection System performed as designed. The licensee's investigation of RPS performance was appropriate.

e. Anticipated Transient Without Scram (ATWS)

The only ATWS mitigation equipment actuated by the event was the dual RR pump trip actuated by the pressure pulse to the ATWS RR pump trip switches' reference leg. These functioned as designed to trip off the RR pumps on indicated low reactor level. The operators spent the first 20-30 seconds after the RR pump trip and associated alarm confirming that no condition existed which required a scram without a scram occurring, hence no ATWS. The available recordings also confirmed no other ATWS actuations, which is consistent with the personnel interviews and the initial valving error, in that, no other ATWS actuations would be expected.

C. Operator Performance

On March 17, 1988, as part of the AIT charter, members of the inspection team interviewed the on shift operators and supervisors who had been assigned to Unit 2 during the March 9, 1988, reactor power oscillation event. The licensee's view of operator actions is included in Attachment 5.

Personnel in the control room at the time of the event included the IM personnel performing the surveillance on a headset with the technician at the instrument rack, an NSO at each unit, the Center Desk NSO (who operates Unit 1 and Unit 2 common systems) and the Shift Control Room Engineer (SCRE). The first indication of a problem was a hi level alarm (due to the initial equalization valving error). This brought the Center Desk Operator (CDO) over to the feedwater control station to assist the Unit 2 NSO. As more alarms came in (low reactor level and 1/2 scram due to the second valving error and resultant pressure pulse) the Unit 2 NSO stationed himself at the RR panel. Based on available indication and the fact that an instrument surveillance was in progress, the operators suspected an instrument problem. The Unit 2 NSO determined that he is zero when both reactor recirculation pumps are tripped. Therefore, the scram setpoint (S) during the instability event was 51%.

Although peak APRM readings reached 118%, the peak output of the thermal power monitor during the instability event was approximately 42% to 45% because of the effect of the R-C circuit on the APRM signal. Therefore, the flow-biased scram never reached its setpoint of 51% and the reactor tripped on high neutron flux at 118%.

d. Reactor Protection System (RPS)

From a review of the alarm printer data, the STARTREC information and interviews with operating personnel, the AIT verified that the only RPS actuation setpoints exceeded were those associated with the initial valving errors and related pressure pulsing of the reference leg which started the event and the APRM hi flux scram which ended the event.

The RPS instrumentation which shares the reference leg with 2B21-N037BB consists of Level 3 (+12.5 inches setpoint), Level 2 (-50 inches setpoint), or Level 1 (-129 inches setpoint) differential pressure indicators and straight pressure indicators. The pressure pulse caused by the valving error is believed to have caused the instruments to see about -40 inches reactor level (or a pulse equal to about 80 inches of water column). This pulse would equal about 2.9 psig which would be too small to affect any of the pressure instruments significantly. This corresponds to the alarm printout which indicates no pressure instrument actuations. The 40 inches level would also be expected to be too small to have affected the Level 1 (-129) or Level 2 (-50) instruments, but would have been expected to actuate the Level 3 instruments. No Level 1 actuations were indicated by the alarm printer. However, the ATWS RR pump trip switches 2B21-N036C and 2B21-N036D which did trip the RR pumps are Level 2 switches. This indicates the pulse was either larger than able to be seen by the level transmitter feeding STARTREC or the switch trip setpoints were conservatively high. Consequently, other Level 2 actuations may or may not have occurred depending on setpoint and the strength of the pulse as seen by the instruments. All of the Level 3 switches actuated as would be expected, including the RPS channel B1 low level alarm and 1/2 scram and the Automatic Depressurization System confirmatory alarm. Both of these are confirmed by the alarm printout.

With regard to the APRM hi flux scram, the licensee's On Site Review (OSR) had concluded (from the alarm printout) had lost RR flow and that ATWS alarms were indicated. At that point, efforts were concentrated on ensuring that a valid ATWS event was not occurring.

The Station Control Room Engineer (SCRE) responded to the initial alarms and, remaining cognizant of his position as the SCRE, positioned himself such that he could clearly observe operator actions and reactor parameters. He then contacted the Shift Engineer (SE).

By this point, the NSOs had determined a valid ATWS did not exist and had directed the IM personnel to stop their surveillance. Multiple feedwater heater alarms were occurring and many of the heaters were isolating extraction steam due to the rapid power reduction. A shift foreman had been sent to the local heater control panels to help restore feedwater heating. At about the time of the Shift Engineer's arrival in the control room, preparations were being made to attempt to restart the RR pumps and power oscillations of 25% magnitude ranging from 25% to 50% began to be seen on the APRM recorders. The SE, upon arriving in the control room, quickly assessed the status of the unit, reviewed the operators' actions, and assumed responsibility for efforts to recover from the reactor power oscillations. At this point, operators were uncertain of the specific actions they should be taking once the power oscillations were observed. The procedures and training appear to have been inadequate. The operators all recognized that they were in the region of core instability, but were uncertain of what actions should be taken. The procedures basically instruct the operator to leave the region of instability without directing "how". Thoughts included; (1) driving in control rods in sequence, (2) using the "CRAM" array of control rods, (3) starting a RR pump, and (4) scram the plant. Operators felt that driving in control rods in sequence would have been too slow in that the first rods to go in would be rods of little reactivity worth. Use of the "CRAM" array of control rods was addressed only in the licensee's procedures pertaining to a loss of feedwater heating consequently operators were uncertain if they should use the "CRAM" array in this condition of 2 RR pump trip and loss of feedwater heating. The operators did not want to scram the plant if they did not have to. Therefore, they decided to try and leave the instability region by restarting a RR pump (either a flow increase or a lowering of the rod line would get the reactor out of the unstable region) which they may have accomplished if the RR pump start sequence could have been satisfied. It is also likely that if a RR pump had been successfully started, the reactor may have scrammed anyway on the resultant power increase. As the operators were unsuccessful in starting a RR pump and were preparing for a manual scram, the reactor scrammed automatically.

The AIT evaluated the response of the individuals versus the information they had available, and has the following observations.

- With the exception of the personnel error by the instrument mechanic (IM) that started the transient, the AIT has concluded that the personnel on-shift at the time of the event took prompt corrective action in accordance with the procedures and training available to them. The AIT believes that the operators could have taken better corrective action by driving in control rods, but that this direction was not provided by procedures or training.

- The procedures available to the operators at the time of the event appear to be inadequate in that they don't specifically address the necessary action to be taken in order to exit the region of instability. General Electric Company's Service Information Letter (SIL) No. 380, issued February 1984, addressed the issue of core stability and the actions to be taken if there were a RR pump(s) trip and the core entered the region of instability. The GE SIL had not been incorporated into the licensee's procedures or training. This issue is further discussed in paragraph III.D.

D. Procedure Adequacy and Training

1. Technical Specifications (see Attachment 4)

The LaSalle Unit 2 Technical Specification (TS) 3.4.1.1.b contained the requirements for loss of both RR pumps or "no reactor coolant system recirculation loops in operation". The TS states "immediately initiate measures to place the unit in at least HOT SHUTDOWN within the next 6 hours." The defined actions, however, were not prescriptive for the seven minutes in which the operators had to act for this event. The TS for loss of a single RR pump 3.4.1.1.a, in step 3.4.1.1.a.2a) 2) discusses increasing core flow. While the reactor was not in this condition (both RR loops were not operating versus one), the operators were aware of this TS, the associated surveillance requirements and the fact that increasing flow would get the reactor out of the unstable region of the power to flow map. These considerations in their own minds served to back up the operators decision to restart RR pumps. The NRR representatives on the AIT were familiar with the development of the LaSalle TS and indicated that part of the reason that LaSalle did not have more specific TS was that the Unit 2 decay factor was submitted as 0.60. With this much margin to a Limit Cycle Oscillation decay factor of 1.0, it was believed that the probability of oscillations at LaSalle would be very low. Since oscillations have now been observed and the decay factor calculation is in question (see Concerns paragraph IV), the AIT believes that the TS were inadequate.

2. GE SIL 380 Revision 1 (Attachment 7)

In 1984, GE issued SIL 380 Revision 1 containing recommendations regarding BWR Core Thermal Hydraulic Stability. This document discusses RR pump trip and recommends, "Immediately reduce power by inserting control rods to or below the 80% rod line using the plant's prescribed control rod shutdown insertion sequence." Further, the SIL recommends that the operation of restarting recirculation pumps should be performed from below the 80% rod line. These recommendations were not incorporated into LaSalle's TS or abnormal operating procedures. Like with TS, part of the motivation to not incorporate these recommendations into procedures was based on the margin to instability provided by the calculated decay factor as well as a skeptical attitude regarding the susceptibility of the LaSalle reactors to instability.

3. Abnormal Procedures and Training

The LaSalle abnormal operating procedures (LOAs) and the associated surveillance procedures (LOS) and general procedures (LGP) provided only general guidance regarding what to do in the event of loss of single or two loop recirculation flow. While guidance and training had been provided to all the licensed operators on how to recognize the onset of instability, there was little guidance provided on what to do next, other than performance of the TS required surveillance and to leave the region of likely instability. The loss of recirc flow procedures were directed at restarting RR pumps. The AIT concluded that the operators reacted as their training and procedures led them, in that, they recognized very well the onset of instability, however, interviews indicated they were confused and uncertain as to how to exit the region. The AIT believes that both the abnormal procedures and the training were inadequate in this regard.

4. Simulator Training

The licensee attempted to simulate LaSalle's instability event on the simulator, but was unsuccessful. The event could possibly be simulated with the instructor interacting with the computer, however to produce an accurate, detailed simulation a new program must be written. Currently, the Production Training Software Group has the data from the actual event and is reviewing possible changes. The licensee stated that they expect a permanent program will probably be available by July 1988 and they will keep the resident informed on the status of the changes.

E. <u>Reporting</u>

As the nature and magnitude of the core power oscillations on LaSalle Unit 2 became widely disseminated and understood such that the AIT was formed on March 16, 1988, several questions were raised regarding the adequacy of the licensee's reporting of this event. These questions were in large part motivated by the fact that the majority of the NRC staff did not learn of the core power oscillations aspect of the event until issuance of the Region III morning report update on March 15, 1988, even though the event had occurred on the evening of March 9, 1988. The AIT examined the reporting aspects of this event through interviews, individual recollections and available tape of phone conversations. The following summarizes the team's observations.

1. Reporting Sequence

   (a) Following the trip of the RR pumps and the subsequent scram at about 5:39 p.m. on March 9, 1988, the licensee's initial investigation classified the event as reportable to the NRC within 4 hours under the provisions of 10 CFR 50.72 due to actuation of the Reactor Protection System (RPS). The licensee actually notified the NRC within 1 hour at about 6:32 p.m. on March 9, 1988. The initial phone call communicated information on the RR pump loss due to personnel error, loss of feedwater heating and reactor trip on high APRM flux. No mention was made of the core power oscillations, however, the licensee's caller stated, "we are still investigating all our alarm typers and everything else." Comments were made to the AIT that the licensee's caller sounded almost jovial over the phone. After listening to the tape of the 50.72 call, this observation was explained to the AIT's satisfaction in that the NRC Headquarter's Duty Officer (HDO) and the licensee's caller recognized each other's voices from previous association and exchanged pleasantries. This situation did not appear to detract from proper communication of information.

   (b) The HDO contacted the Regional Duty Officer (RDO) at 7:21 p.m. CST, who, in turn, notified the Cognizant Section Chief (also AIT leader) and Branch Chief.

   (c) At approximately 7:00 p.m. CST on March 9, 1988, the licensee and the Resident Inspector (RI) succeeded in establishing contact and the RI was briefed regarding the event. The RI was offsite attending the Region III Resident Seminar (March 8-10). The licensee informed the RI that the APRMs had oscillated between 25-50% during the event and that the reactor had scrammed on APRM high flux at 118% setpoint. At this point, the RI did not appreciate a significant concern with the APRM oscillation in that oscillations are a potentially expected phenomena given the parameters of high reactor core power and low reactor core flow.

(d) On March 10, 1988, at approximately 10:00 a.m., Region III issued a morning report describing the event. No mention of oscillations was made in this report. The high power scram was attributed to a power surge caused by the cold feedwater.

(e) At approximately 12:30 p.m. on March 10, 1988, the Region III Section Chief contacted the LaSalle Station Manager for the purpose of ensuring the information in a Preliminary Notification (PN) was correct. During this conversation, the Station Manager pointed out that while the PN was substantially correct, the unit had experienced some neutron flux oscillations during the event which were being evaluated. The Region III Section Chief did not include any information on the oscillations in the PN based on the evaluation still being in progress.

(f) On March 11, 1988, the RI returned to the site and participated in several discussions with the licensee involving the event including a meeting where the STARTREC traces showing the oscillations (20%-95%) were made available. The RI was informed that the licensee had involved GE in the evaluation and that GE had indicated the oscillations were within analysis bounds and provided for by the APRM high flux scram. Also, on March 11, 1988, the Station Manager contacted the Region III Section Chief and provided additional information including an explanation of the reactor's position on the power to flow map, the effect of the loss of feedwater heating, and the onset of flux oscillations. During this call, the Section Chief does not recall that the magnitude of the oscillations was discussed, but, rather the Station Manager indicated the oscillations were being evaluated and the reactor vendor was involved. The Station Manager indicated that GE felt the oscillations were within analysis bounds, that the core performed as expected for the conditions and he offered to make a presentation on the event to Region III if desired. The Section Chief indicated this may be a good idea but wanted to wait until the RI and possibly Region III core physics inspectors had a chance to look at the data. Following his review of the available information, the RI contacted the Section Chief on March 11, 1988, and communicated the oscillation information which indicated the event was more severe than initially believed. At this point, the Region III Branch Chief and Division Director were informed that flux oscillations had occurred but were within analysis and a more complete description would be available on Monday, March 14 following the RI's further investigation. The NRR Projects Office was similarly informed.

(g) On March 14, 1988, the RI was given a copy of CECo's Potentially Significant Event (PSE) report and an advance copy of a Rapid SIL generated by GE addressing the event. Due to following up on additional information on the event and the length and complexity of the morning report, the RI was unable to get the report documented and transmitted by the 10:00 a.m. deadline on March 14, 1988. Since Region III was waiting for further information from the site, no additional actions were taken by the Region on March 14, 1988.

(h) On March 15, 1988, the Region III morning report update describing the event, including the oscillations, was disseminated, resulting in the formation of the AIT on March 16, 1988.

2. Reporting Evaluation

In the AIT's view, the licensee correctly classified the event on the initial call on March 9, 1988, as being 4 hour reportable based on the requirements of 10 CFR 50.72 for an RPS actuation. While the initial call did not mention oscillations, the caller stated that the information was still being evaluated. Consequently, the AIT believes the initial call was satisfactory for as far as it went. However, 10 CFR 50.72(c) requires the licensee to make followup notifications in addition to initial notifications as follows:

> (2) Immediately report (i) the results of ensuing evaluations or assessments of plant conditions, (ii) the effectiveness of response or protective measures taken, and (iii) information related to plant behavior that is not understood.

The licensee made no formal followup reports via the ENS circuit (red phone) to the HDO. The AIT believes that such a followup call would have been appropriate since the occurrence of oscillations and the nature and magnitude of the oscillations would significantly add to the information communicated on the initial call. The AIT believes the licensee felt that subsequent calls to the RI and the Region III Section Chief would accomplish the intent of communicating followup information. The AIT believes this to be incorrect for two reasons. First, the intent of 10 CFR 50.72 is to describe the formal notification process which is to be conducted over the ENS circuit and notification of other NRC personnel (including the RI) does not relieve the licensee of the responsibility to notify the NRC via ENS. Second, the information communicated to the NRC in the subsequent calls was insufficient to allow the NRC to appreciate the nature and magnitude of the oscillations. The NRC has a need for prompt, pertinent information on this type event which is at least equivalent to the information which the licensee furnished to GE.

IV. AIT CONCERNS AND RECOMMENDATIONS

A. Concerns

The following paragraphs document concerns developed by the AIT during the review of this event. Several of these concerns were discussed with the licensee prior to the exit and the licensee has provided several responses in Attachment 5 to this report. These responses had not been completely evaluated at the time of the writing of this report. Other issues may require further study and subsequent response.

1. Decay Ratio

Decay ratio in a reactor is a measure of the response of the neutron flux to a change or perturbation. A decay ratio of less than 1.0 indicates that the response to a perturbation will decay to the steady state value (see Attachment 3, Figures 2, 3 and 4). A decay ratio of 1.0 represents the special condition when the response to a perturbation will be a steady state oscillation of constant magnitude (limit cycles). GE predictive methods for determining BWR decay ratio were approved with 20% uncertainty. Predicted decay ratio for LaSalle conditions was 0.60 (40% uncertainty) yet limit cycles were observed. The only obvious deviation from prediction assumptions was in core water level and associated natural circulation flow rate. The 40% difference needs to be explained. Have calculations been discredited as acceptable evidence of core stability?

2. Technical Specifications (TS)

BWR thermal hydraulic stability was the subject of Generic Issue B-19. Early BWRs were designed to maintain decay ratios less than 0.5 (a decay ratio = 1.0 corresponds to an undamped limit cycle oscillation). Later core designs tended towards decay ratios of 1.0 due to higher power density cores and changing fuel design characteristics.

In the resolution of Generic Issue B-19, core designs which were potentially unstable (DR = 1.0) under natural circulation operating conditions were approved with the provision that operating procedures and technical specifications would assure that neutron flux oscillations indicative of core instability would be readily detected and suppressed as required by GDC 12. Licensees were informed by Generic Letter 86-02 that such procedures and TS must be implemented for new reload cores unless it could be demonstrated by approved calculation methods that the core was stable throughout permissible operating regions of the power/flow map. Calculated core decay ratios of less than 0.80 by General Electric methods were approved as acceptable evidence of core stability.

LaSalle Unit 2 did not have fully implemented procedures and TS in accordance with the B-19 resolution because the calculated decay ratio for the current operating cycle is 0.60. LaSalle Unit 2 did have TS and procedures for stability surveillance under conditions of single loop operation (SLO), but had declined suggestions by the NRC staff that these should also apply to two loop operation because higher decay ratios were certain for future reloads. The licensee has been informed that revised TS are required for Unit 2 now that oscillations have actually been observed.

LaSalle Unit 1 is currently under review for reload. The licensee has been informed that stability technical specifications will be required for that reactor even though the calculated decay ratio is 0.75 (indicative that it is less stable than Unit 2).

Since several BWRs have higher decay ratios than LaSalle, the issue of decay ratio in paragraph IV.A and stability TS is considered generic. Improved standard technical specifications relating to stability need to be developed. Further, the implementation status of stability TS on all BWRs should be reviewed with particular attention to the adequacy of requirements for response to loss of RR pumps. The criteria for applicability of stability TS based on decay ratio calculations and reactor type should be reexamined.

3. Instrumentation

The following concerns with the available instrumentation at LaSalle were developed by the AIT:

- The slow pen response time and chart speed of the APRM strip chart recorders in the control room reduced indicated amplitude of oscillations.
- Time delay relays for the LPRM Hi and APRM Hi alarms delay recognition.
- Running Average of APRM signal in Power/Flow circuit delays or prevents reactor trip at lower power levels with reactor oscillations.
- If oscillations are regional in nature and LPRM signals are out of phase, LPRM inputs to APRMs will tend to cancel each other so that oscillations are not evident on the APRMs.
- Hi Speed Recording of APRM signals used for event evaluation was triggered by Low Water Level signal - not always a condition of the instability event. This data

is needed to assess the nature and magnitude of neutron flux oscillations and the safety of restart after an instability event.

- LaSalle and some other BWRs do not have high speed data recording instrumentation which can be committed for availability during plant operation.

4. Oscillation Characteristics

Some characteristics of the LaSalle neutron flux oscillations were atypical of previous events and have led to concerns about the applicability of previous safety analyses. The magnitude of in-phase limit cycle oscillations previously observed on the APRMs during special stability tests and operating reactor events were typically in the range of 5% to 15% (peak-to-peak) of rated power, and as high as 25%. This compares to peak-to-peak values of about 100% at the time of the 118% neutron flux trip for LaSalle.

The estimated value of local power at the time of trip was greater than 310% and LPRM readings indicate that the core power peak shifted and increased by 25%. Even though the fuel LHGR limit of 13.4 kw/ft was not exceeded because of the thermal time constant of the fuel, the increased power peaking was unexpected based on Vermont Yankee stability tests, and was not factored into the generic safety evaluation performed by GE during review of the thermal hydraulic stability Generic Issue B-19.

The previous GE safety analyses considered several limiting moderate frequency transients which were initiated while the neutron flux was oscillating below the 120% scram setpoint, and included a rod withdrawal error with the flux oscillating up to the 120% scram level. Additional analyses were performed to evaluate the impact of oscillations that approached 300% of rated neutron flux (e.g., regional oscillations) without scram prior to rod insertion and termination of the event. All of these analyses showed that significant fuel thermal margin existed to safety limits. While there are several aspects of these analyses which differ from LaSalle (initial power level and amplitude of the oscillations; no change in bundle peaking factors due to the event, etc.), the AIT agrees that they are sufficiently representative and conservative to demonstrate that no fuel thermal or mechanical limits were exceeded during the event. However, reliable detection and suppression provisions are necessary to assure protection against future events which could involve regional oscillations to higher power levels.

The licensee was also asked to review the impact of the event on stability considerations addressed in the 1979 GE Generic ATWS report, "Assessment of BWR Mitigation of ATWS" (NEDE-24222).

The report does specifically investigate the sensitivity and potential impact of limit cycle neutron flux oscillations up to 500% of rated bundle power following recirculation pump trip. It was concluded that oscillations of this magnitude would not result in sufficient fuel clad temperature variation (130°F) to affect fuel integrity. It was further concluded that a loss of clad integrity due to prolonged exposure to limit cycles was an acceptable consequence in view of the importance of the recirculation pump trip (RPT) to minimize the energy deposited in the suppression pool (thereby maintaining containment pressure within limits) during an ATWS event.

In view of the large magnitude of the APRM oscillations in LaSalle, the AIT believes that the ultimate power level without scram is unknown, and that the 500% level assumed in the ATWS investigation may not be bounding. LPRM oscillation magnitudes more than seven times those of the APRMs have been observed in the case of regional oscillations. The licensee reports that the BWROG is discussing this issue (inherent power limits) and the licensee will provide a status report on July 1, 1988.

5. Additional Concerns

Several additional concerns were presented to the licensee in the form of questions. These questions and the licensee's response are contained in Attachment 5 to this report.

B. Recommendations

The AIT recommends that the concerns identified in items IV.A.1 through IV.A.5 of this report be examined by NRR for generic and [illegible] specific resolution. In the interim, the AIT recommends that revised stability TS as discussed in IV.A.2 be developed for LaSalle Units 1 and 2 and the licensee be authorizied via letter to modify interim operating procedures provided they remain consistent with the new T.S. The revised technical specifications and procedures should incorporate the changes summarized in Attachment 5 (Appendix A, Item 3), which include immediate insertion of high worth rods and observation of APRM/LPRM noise when no pumps are operating and power is above the 80% Rod Control Line. The reactor is to be tripped immediately whenever instability is suspected. It is expected that the time available (greater than 5 minutes) to instability following a two pump trip transient is sufficient to permit manual power reduction, avoiding the need for reactor trip unless the core is unstable by a large margin. Proposed procedures permit manual action for up to two minutes (prior to scram) to reverse operating actions which may result in small margins of instability when one or both pumps are operating.

V. AIT CONCLUSIONS

The AIT finds that the core power oscillations observed on LaSalle Unit 2 on March 9, 1988, were initiated by a personnel error resulting in the trip of both recirculation pumps compounded by the loss of steam heating to the feedwater. The licensee's evaluation of the event (including assistance from the reactor vendor) is still continuing, however, to date is believed to have been thorough and proper. No evidence of any fuel damage has been noted related to the event. Performance of equipment during the event is believed to have been as designed for the conditions encountered during the event. Operator performance, while it could have been better, is believed to have been prompt, appropriately controlled and within the bounds of the procedures and training available to them. Both the procedures and training available to the operators are believed to have been inadequate in that prudent recommendations of GE SIL 380 were not incorporated and little direction was provided regarding what to do in the event of instability. Reporting is believed to have been inadequate in that no formal followup report was made regarding the results of the investigation and determination of flux oscillations. The licensee's response to NRC initiatives by shutting down, providing the AIT with prompt and technically sound information and responding to the CAL is believed to have been excellent. Several concerns and recommendations on this event remain to be followed as documented in paragraph IV.

On March 17, 1988, the AIT determined that the licensee had complied with the restart provisions of the CAL, including a requirement for a manual scram in the event of a loss of both recirculation pumps from an operating condition, and the Region III Administrator's designee authorized restart of LaSalle Unit 2 at 10:45 p.m. The AIT and the licensee have agreed on the essential features of revised technical specifications and operating procedures to protect against this event during future operation.

VI. EXIT INTERVIEW

The inspectors met with licensee representatives (denoted in paragraph I.D.) informally throughout the inspection period and at the conclusion of the onsite inspection activities on March 24, 1988, and summarized the scope and findings of the inspection activities.

The inspectors also discussed the likely informational content of the inspection report with regard to documents or processes reviewed by the inspectors during the inspection. While the inspectors did review some proprietary material, none of the areas expected to be contained in the report were identified by the licensee as proprietary. The licensee acknowledged the findings of the inspection.

CAL-RIII-88-03

Docket No. 50-374

Commonwealth Edison Company
ATTN: Mr. Cordell Reed
Senior Vice President
Post Office Box 767
Chicago, IL 60690

Gentlemen:

This letter confirms the telephone conversation between E. G. Greenman of this office and you on March 17, 1988, related to the dual recirculation pump trip and related core performance anomalies occurring at LaSalle Unit 2 on March 9, 1988. With regard to this event, we understand that you will:

1. Perform an evaluation of reactor performance during this event including secondary systems, the reactor protection system, and ATWS systems.

2. Perform an evaluation of operator performance during this event.

3. Evaluate the adequacy of your Technical Specifications, operating procedures, abnormal operating procedures, and emergency procedures with respect to this event and vendor recommendations (GE SIL-380).

4. Perform increased activity level sampling during Unit 2 startup to verify no abnormalities.

5. Submit to NRC Region III a formal report of your findings and conclusions within 30 days of receipt of this letter.

Throughout this investigative effort, we understand that you will take those actions necessary to ensure that complete documentary evidence of the conditions being examined is maintained, and furnished to the NRC's Augmented Inspection Team which was initiated on March 16, 1988.

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Attachment 1

We further understand that startup of Unit 2 will occur after a determination has been made by the AIT Team Leader that equipment performance was satisfactory, including verification of reactor coolant samples, and that interim procedural changes are satisfactory including a requirement to trip the reactor if no reactor recirculation loops are in operation, and the reactor is in operational conditions 1 or 2, and with concurrence of the Regional Administrator or his designee. Issuance of this Confirmatory Action Letter does not preclude the issuance of an order requiring implementation of the above commitments.

None of the actions specified herein should be construed to take precedence over actions which you feel necessary to ensure plant and personnel safety.

Please advise us immediately if your understanding differs from that set forth above.

Sincerely,

Orignial Signed by A. Bert Davis

A. Bert Davis
Regional Administrator

cc: D. Butterfield, Nuclear Licensing Manager
G. J. Diederich, Plant Manager
DCD/DCB (RIDS)
Licensing Fee Management Branch
Resident Inspector, RIII
Richard Hubbard
J. W. McCaffrey, Chief, Public Utilities Division
David Rosenblatt, Governor's Office of Consumer Services
J. M. Taylor, DEDO
T. E. Murley, NRR
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W. Lanning, NRR
F. Miraglia, NRR
G. Holahan, NRR
D. Crutchfield, NRR
J. Partlow, NRR
J. Strasma, RIII
J. Goldberg, OGC
D. Muller, NRR