10CFR50.59 Safety Evaluation Cover Sheet

Station LaSalle
Modification/Minor Plant Change # M1-1-90-009
M1-2-90-007

Design Issues Worksheets have been completed prior to Safety Evaluation. The following design issues could impact the Safety Evaluation and should be considered during performance of the Safety Evaluation, particularly during Steps 5 (normal operation) and 6 (failure modes):

M12, Seismic Qualification
M13, Design Loads
OP6, TIP Tube Supports & LPRM/SRM Cables
R9, Radiation Exposure
ST3, Structural Integrity

[ ] This evaluation identified an Unreviewed Safety Question. See Item 14 on the 10CFR50.59 Safety Evaluation form.

[X] A Technical Specification change is required and a Technical Specification Revision Request has been prepared. See Item 14 on the 10CFR50.59 Safety Evaluation form.

[ ] This evaluation did not identify an Unreviewed Safety Question and no Technical Specification change is required. The modification or minor plant change may be installed without prior NRC approval.

AH 9-14-92 MMM 9/15/92

DM 9-14-92

K. R. Kehring Date 9/15/92
Cognizant Engineer

J D Williams Date 9/15/92
Design Superintendent or Supervisor

Exhibit D
ENC-QE-06.1
Revision 5

Technical Specification Revisions for Modification

Station LaSalle

Unit(s) 1&2

Modification # M1-1-90-009
M1-2-90-007

To: ______________ (Systems Design Superintendent)
______________ (NLA)

______________ (Station Regulatory Assurance Supervisor)

List required Technical Specification revisions:

Section 3/4.1.3 Bases.

Recommend effective date for revision (i.e., calendar date, beginning of outage #, or end of outage #)

AH 9-14-92
DMM 9-14-92
MJm 9-15-92

Prepared by: J D Welle Date: 9/15/92

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Exhibit E
10CFR50.59 SAFETY EVALUATION

1. List the documents implementing the proposed change.

   ECN 01-00169M, ECN 01-00170M

2. Describe the proposed change and the reason for the change.

   The proposed change increases the nominal clearance between the Control Rod Drive (CRD) Housing and the CRD Support Structure (Shoot-out Steel) from 1 inch to 1.5 inches at ambient temperature by lowering the support structure. This change facilitates undervessel work, thereby reducing radiation exposure to plant personnel.

3. Is the change:

   [X] Permanent

   [ ] Temporary -
   Expected duration

   AND
   Plant Mode(s) restrictions while installed NO
   (NONE if no plant mode restrictions apply)

4. List the SAR sections which describe the affected systems, structures, or components (SSCs) or activities. Also list the SAR accident analysis sections which discuss the affected SSCs or their operation. List any other controlling documents such as SERs, previous modifications or Safety Evaluations, etc.

   4.6.1.2, 4.6.2.3, 15.4.1, 15.4.2, 15.4.3, 15.4.8

5. Describe how the change will affect plant operation when the changed SSCs function as intended (i.e., focus on system operation/interactions in the absence of equipment failures). Consider all applicable operating modes. Include a discussion of any changed interactions with other SSCs.

   No effect on plant operation. Refer to attached GE safety evaluation document B13-01503, Rev. 1, dated September 92.

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6. Describe how the change will affect equipment failures. In particular, describe any new failure modes and their impact during all applicable operating modes.

   No effect on equipment failures. Refer to attached GE safety evaluation document B13-01503, Rev. 1, dated September 92.

7. Identify each accident or anticipated transient (i.e., large/small break LOCA, loss of load, turbine missiles, fire, flooding) described in the SAR where any of the following is true:
   - The change alters the initial conditions used in the SAR analysis
   - The changed SSC is explicitly or implicitly assumed to function during or after the accident
   - Operation or failure of the changed SSC could lead to the accident

ACCIDENT	SAR SECTION
Rod Ejection Accidents	4.6.1.2, 4.6.2.3, 15.4.8

8. List each Technical Specification (Safety Limit, Limiting Safety System Setting or Limiting Condition for Operation) where the requirement, associated action items, associated surveillances, or bases may be affected. To determine the factors affecting the specification, it is necessary to review the FSAR and SER where the bases section of the Technical Specifications does not explicitely state the basis.

   Section 3/4.1.3 Bases

9. Will the change involve a Technical Specification revision?

   [X] Yes [ ] No

   If a Technical Specification revision is involved, the change cannot be implemented until the NRC issues a license amendment. When completing Step 14, indicate that a Technical Specification revision is required.

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10. To determine if the probability or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the SAR may be increased, use one copy of this page to answer the following questions for each accident listed in Step 7. Provide the rationale for all NO answers.

Affected accident
Rod Ejection
Accidents

SAR Section:
4.6.1.2,
4.6.2.3,
15.4.8

May the probability of the accident be increased? [ ] Yes [X] No

See attached GE safety evaluation document B13-01503, Rev. 1, dated September 92.

May the consequences of the accident (off-site dose) be increased? [ ] Yes [X] No

See attached GE safety evaluation document B13-01503, Rev. 1, dated September 92.

May the probability of a malfunction of equipment important to safety increase? [ ] Yes [X] No

See attached GE safety evaluation document B13-01503, Rev. 1, dated September 92.

May the consequences of a malfunction of equipment important to safety increase? [ ] Yes [X] No

See attached GE safety evaluation document B13-01503, Rev. 1, dated September 92.

If any answer to Question 10 is YES, then an Unreviewed Safety Question exists.

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11. Based on your answers to Questions 5 and 6, does the change adversely impact systems or functions so as to create the possibility of an accident or malfunction of a type different from those evaluated in the SAR?

[ ] Yes [X] No

Describe the rationale for your answer.

See attached GE safety evaluation document B13-0'503, Rev. 1, dated September 92.

If the answer to Question 11 is Yes, then an Unreviewed Safety Question exists.

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12. Determine if parameters used to establish the Technical Specification limits are changed. Use one copy of this page to answer the following questions for each Technical Specification listed in Step 8. List the Technical Specification, Technical Specification Bases, SAR and SER Sections reviewed for this evaluation.

Tech. Specs. 3/4.1.3 and Bases, UFSAR Sec. 4.6.1.2, 4.6.2.2 and 15.4.e

Evaluation of Technical Specification
(Enter N/A if none are affected and check last option.)

3/4.1.3 Bases

(Check appropriate condition):

[ ] All changes to the parameters or conditions used to establish the Technical Specification requirements are in a conservative direction. Therefore, the actual acceptance limit need not be identified to determine that no reduction in margin of safety exists - proceed to Question 13.

[ ] The Technical Specification or SAR provides a margin of safety or acceptance limit for the applicable parameter or condition. List the limit(s)/margin(s) and applicable reference for the margin of safety below - proceed to question 13.

[X] The applicable parameter or condition change is in a potentially non-conservative direction and neither the Technical Specification, the SAR, or the SER provides a margin of safety or an acceptance limit. Request Nuclear Licensing assistance to identify the acceptance limit/margin for the Margin of Safety determination by consulting the NRC, SAR, SER's or other appropriate references. List the agreed limit(s)/margin(s) below.

[ ] The change does not affect any parameters upon which Technical Specifications are based; therefore, there is no reduction in the margin of safety. Proceed to question 14.

List Acceptance Limit(s)/Margin(s) of Safety

Control rod movement must be less than 6 inches (one drive notch) to be bounded by current FSAR analysis. See UFSAR Section 4.6.2.3.3.1 and attached GE safety evaluation document B13-01503, Rev. 1, dated September 92.

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Tech Spec 3/4.1.3.8
Tech Spec and
Tech Spec 3/4.1.3
Tech Spec Bases

SAR Section 4.6.1.2,
SAR Section 4.6.2.3,
SAR Section 15.4.8

13. Use the above limits to determine if the margin of safety is reduced (i.e., the new values exceed the acceptance limits). Describe the rationale for your determination. Include a description of compensating factors used to reach that conclusion.

See attached GE safety evaluation document B13-01503, Rev. 1, dated September 92.

If a Margin of Safety is reduced an Unreviewed Safety Question exists.

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14. Check one of the following:

[ ] An Unreviewed Safety Question was identified in Step 10, Step 11, or Step 13. The proposed change MUST NOT be implemented without NRC approval.

[ ] No Unreviewed Safety Question will result (Steps 10, 11, and 13) AND no Technical Specification revision will be involved. The change may be implemented in accordance with applicable procedures.

[X] A Technical Specification revision is involved; but no Unreviewed Safety Question will result. The proposed change requires a License Amendment. Notify Station Regulatory Assurance and Nuclear Licensing that a Technical Specification revision is required. Mark below as applicable.

[ ] The change is not a plant modification or minor plant change and will not be implemented under 10CFR50.59. Upon receipt of the approved Technical Specification change from the NRC, the change may be implemented.

[X] The change is a plant modification or minor plant change. Mark below as applicable.

[X] A revision to an existing Technical Specification is required. The change MUST NOT be installed until receipt of an approved Technical Specification revision.

[ ] The change will not conflict with any existing Technical Specifications and only new Technical Specifications are required. In these cases, Nuclear Licensing may authorize installation, but not operation, prior to receipt of NRC approval of the License Amendment. If such authorization is granted, the block below should be checked.

[ ] Nuclear Licensing has authorized installation, but not operation, prior to receipt of NRC approval of the License Amendment. The 10CFR50.59 Safety Evaluation indicates that no Unreviewed Safety Question will result and provides authority for installation only.

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Note: Partial Modifications and/or separate 10CFR50.59 reviews for portions of the work may be used to facilitate installation.

Preparer K.R. Kehring 9/15/92
(Cognizant Engineer) Date

15. The reviewer has determined that the documentation is adequate to support the above conclusion and agrees with the conclusion.

Reviewer J D Williams 9/15/92
(Design Superintendent/Supervisor) Date

4.6.2.3.1.2 Analysis of Malfunction Relating to Rod Withdrawal

There are no known single malfunctions that cause the unplanned withdrawal of even a single control rod; providing initiating signal has not been given (Subsections 4.6.1.1.1.1.1, Item c, part 1, and 4.6.2.3.1.2.10). However, if multiple malfunctions are postulated, studies show that an unplanned rod withdrawal can occur at withdrawal speeds that vary with the combination of malfunctions postulated. In all cases the subsequent withdrawal speeds are less than that assumed in the rod drop accident analysis as discussed in Chapter 15.0. Therefore, the physical and radiological consequences of such rod withdrawals are less than those analyzed in the rod drop accident.

4.6.2.3.1.2.1 Drive Housing Fails at Attachment Weld

The bottom head of the reactor vessel has a penetration for each control rod drive location. A drive housing is raised into position inside each penetration and fastened by welding. The drive is raised into the drive housing and bolted to a flange at the bottom of the housing. The housing material is seamless, Type 304 stainless steel pipe with a minimum tensile strength of 75,000 psi. The basic failure considered here is a complete circumferential crack through the housing wall at an elevation just below the J-weld.

Static loads on the housing wall include the weight of the drive and the control rod, the weight of the housing below the J-weld, and the reactor pressure acting on the 6-inch diameter cross-sectional area of the housing and the drive. Dynamic loading results from the reaction force during drive operation.

If the housing were to fail as described, the following sequence of events is foreseen. The housing would separate from the vessel. The control rod, drive, and housing would be blown downward against the support structure by reactor pressure acting on the cross-sectional area of the housing and the drive. The downward motion of the drive and associated parts would be determined by the gap between the bottom of the drive and the support structure and by the deflection of the support structure under load. In the current design, maximum deflection is 3.65 approximately 3 inches. If the collet were to remain latched, no further control rod ejection would occur (Reference 4); the housing would not drop far enough to clear the vessel penetration. Reactor water would leak at a rate of approximately 220 gpm through the 0.03-inch diametral clearance between the housing and the vessel penetration.

If the basic housing failure were to occur while the control rod is being withdrawn (this is a small fraction of the total drive operating time) and if the collet were to stay unlatched, the following sequence of events is foreseen. The housing would separate from the vessel. The drive and housing would be blown downward against the control rod drive housing support.

be slightly less than that for drive housing failure because reactor pressure would act on the drive cross-sectional area only and the housing would remain attached to the reactor vessel. The drive would be isolated from the cooling water supply. Reactor water would flow downward past the velocity limiter piston, through the large drive filter, and into the annular space between the thermal sleeve and the drive. For worst-case leakage calculations, the large filter is assumed to be deformed or swept out of the way so it would offer no significant flow restriction. At a point near the top of the annulus, where pressure would have dropped to 350 psi, the water would flash to steam and cause choke-flow conditions. Steam would flow down the annulus and out the space between the housing and the drive flanges to the atmosphere. Steam formation would limit the leakage rate to approximately 840 gpm.

If the collet were latched, control rod ejection would be limited to the distance the drive can drop before coming to rest on the support structure. There would be no tendency for the collet to unlatch because pressure below the collet piston would drop to zero. Pressure forces, in fact, exert 1435 pounds to hold the collet in the latched position.

If the bolts failed during control rod withdrawal, pressure below the collet piston would drop to zero. The collet, with 1650 pounds return force, would latch and stop rod withdrawal.

4.6.2.3.1.2.4 <u>Weld Joining Flange to Housing Fails in Tension</u>

The failure considered is a crack in or near the weld that joins the flange to the housing. This weld extends through the wall and completely around the housing. The flange material is forged, Type 304 stainless steel, with a minimum tensile strength of 75,000 psi. The housing material is seamless, Type 304 stainless steel pipe, with a minimum tensile strength of 75,000 psi. The conventional, full-penetration weld of Type 308 stainless steel has a minimum tensile strength approximately the same as that for the parent metal. The design pressure and temperature are 1250 psig and 575° F. Reactor pressure acting on the cross-sectional area of the drive, the weight of the control rod, drive, and flange, and the dynamic reaction force during drive operation result in a maximum tensile stress at the weld of approximately 6000 psi.

If the basic flange-to-housing joint failure occurred, the flange and the attached drive would be blown downward against the support structure. The support structure loading would be slightly less than that for drive housing failure because reactor pressure would act only on the drive cross-sectional area. Lack of differential pressure across the collet piston would cause the collet to remain latched and limit control rod motion to 3.63 approximately 3 inches. Downward drive movement would be small and, therefore, most of the drive would remain inside the housing. The pressure-under and pressure-over lines are flexible

4.6.2.3.2.1 Reliability Analysis

A reliability analysis was performed to demonstrate that the ARI design meets the design failure rate criteria of 10^{-6} failures to actuate per reactor-year (reference 5). The probability of spurious actuation was shown to be more than a factor of 10 less likely than the probability of failure to actuate. The basis for demonstrating the 10^{-6} criteria was the complete electrical independence of the ARI system from the electrical portion of the reactor protection system (RPS) including power supplies. When determining the overall electrical system failure probability (ARI and RPS), the independence results in an overall failure probability well beyond any practical means of engineering judgement ($\sim 10^{-11}$ failures to actuate per demand). Note that the mechanical portion of the CRD is unchanged by the ARI modification and now becomes the limiting factor in the overall scram system reliability. Hence, the ARI modification provides a conservative means of demonstrating adequate ATWS prevention for the expected ATWS initiators.

The charging water header pressure is monitored with a low pressure alarm to provide warning to control room operators of an impending reactor scram due to low charging-water-header pressure.

The scram assures that sufficient energy remains in the accumulators to shut down the reactor.

4.6.2.3.2.2 Control Rod Support and Operation

As described previously, each control rod is independently supported and controlled as required by safety design bases.

4.6.2.3.3 Control Rod Drive Housing Supports

4.6.2.3.3.1 Safety Evaluation

Downward travel of the CRD housing and its control rod following the postulated housing failure equals the sum of these distances: (1) the compression of the disc springs under dynamic loading, and (2) the initial gap between the grid and the bottom contact surface of the CRD flange. If the reactor were cold and pressurized, the downward motion of the control rod would be limited to the spring compression (approximately 2 inches) plus a gap of approximately 1 inch. If the reactor were hot and pressurized, the gap would be reduced approximately 1/4 inch and the spring compression would be slightly less than in the cold condition. In either case, the control rod movement following a housing failure is substantially limited below one drive notch movement (6 inches). Sudden withdrawal of any control rod through a distance of one drive notch at any position in the core does not produce a transient sufficient to damage any radioactive material barrier.

1-1/2

The CRD housing supports are in place during power operation and when the nuclear system is pressurized. If a control rod is ejected during shutdown, the reactor remains subcritical because it is designed to remain subcritical with any one control rod fully withdrawn at any time.

1-1/4

At plant operating temperature, a gap of approximately 1/4 inch exists between the CRD housing and the supports. At lower temperatures the gap is greater. Because the supports do not contact any of the CRD housing except during the postulated accident condition, vertical contact stresses are prevented.

CONTROL RODS (Continued)

In addition, the automatic CRD charging water header low pressure scram (see Table 2.2.1-1) initiates well before any accumulator loses its full capability to insert the control rod. With this added automatic scram feature, the surveillance of each individual accumulator check valve is no longer necessary to demonstrate adequate stored energy is available for normal scram action.

Control rod coupling integrity is required to ensure compliance with the analysis of the rod drop accident in the FSAR. The overtravel position feature provides the only positive means of determining that a rod is properly coupled and therefore this check must be performed prior to achieving criticality after completing CORE ALTERATIONS that could have affected the control rod drive coupling integrity. The subsequent check is performed as a backup to the initial demonstration.

In order to ensure that the control rod patterns can be followed and therefore that other parameters are within their limits, the control rod position indication system must be OPERABLE.

3.65

The control rod housing support restricts the outward movement of a control rod to less than 2 inches in the event of a housing failure. The amount of rod reactivity which could be added by this small amount of rod withdrawal is less than a normal withdrawal increment and will not contribute to any damage to the primary coolant system. The support is not required when there is no pressure to act as a driving force to rapidly eject a drive housing.

The required surveillance intervals are adequate to determine that the rods are OPERABLE and not so frequent as to cause excessive wear on the system components.

3/4.1.4 CONTROL ROD PROGRAM CONTROLS

Control rod withdrawal and insertion sequences are established to assure that the maximum insequence individual control rod or control rod segments which are withdrawn at any time during the fuel cycle could not be worth enough to result in a peak fuel enthalpy greater than 280 cal/gm in the event of a control rod drop accident. The specified sequences are characterized by homogeneous, scattered patterns of control rod withdrawal. When THERMAL POWER is greater than 20% of RATED THERMAL POWER, there is no possible rod worth which, if dropped at the design rate of the velocity limiter, could result in a peak enthalpy of 280 cal/gm. Thus requiring the RSCS and RWM to be OPERABLE when THERMAL POWER is less than or equal to 20% of RATED THERMAL POWER provides adequate control.

The RSCS and RWM provide automatic supervision to assure that out-of-sequence rods will not be withdrawn or inserted.

The analysis of the rod drop accident is presented in Section 15.4.9 of the FSAR and the techniques of the analysis are presented in a topical report, Reference 1, and two supplements, References 2 and 3.

CONTROL RODS (Continued)

In addition, the automatic CRD charging water header low pressure scram (see Table 2.2.1-1) initiates well before any accumulator loses its full capability to insert the control rod. With the added automatic scram feature, the surveillance of each individual accumulator check valve is no longer necessary to demonstrate adequate stored energy is available for normal scram action.

Control rod coupling integrity is required to ensure compliance with the analysis of the rod drop accident in the FSAR. The overtravel position feature provides the only positive means of determining that a rod is properly coupled and therefore this check must be performed prior to achieving criticality after completing CORE ALTERATIONS that could have affected the control rod drive coupling integrity. The subsequent check is performed as a backup to the initial demonstration.

In order to ensure that the control rod patterns can be followed and therefore that other parameters are within their limits, the control rod position indication system must be OPERABLE. 3.65

The control rod housing support restricts the outward movement of a control rod to less than 8 inches in the event of a housing failure. The amount of rod reactivity which could be added by this small amount of rod withdrawal is less than a normal withdrawal increment and will not contribute to any damage to the primary coolant system. The support is not required when there is no pressure to act as a driving force to rapidly eject a drive housing.

The required surveillance intervals are adequate to determine that the rods are OPERABLE and not so frequent as to cause excessive wear on the system components.

3/4.1.4 CONTROL ROD PROGRAM CONTROLS

Control rod withdrawal and insertion sequences are established to assure that the maximum insequence individual control rod or control rod segments which are withdrawn at any time during the fuel cycle could not be worth enough to result in a peak fuel enthalpy greater than 280 cal/gm in the event of a control rod drop accident. The specified sequences are characterized by homogeneous, scattered patterns of control rod withdrawal. When THERMAL POWER is greater than 20% of RATED THERMAL POWER, there is no possible rod worth which, if dropped at the design rate of the velocity limiter, could result in a peak enthalpy of 280 cal/gm. Thus requiring the RSCS and RWM to be OPERABLE when THERMAL POWER is less than or equal to 20% of RATED THERMAL POWER provides adequate control.

The RSCS and RWM provide automatic supervision to assure that out-of-sequence rods will not be withdrawn or inserted.

The analysis of the rod drop accident is presented in Section 15.4.9 of the FSAR and the techniques of the analysis are presented in a topical report, Reference 1, and two supplements, References 2 and 3.

The RBM is designed to automatically prevent fuel damage in the event of erroneous rod withdrawal from locations of high power density during high power operation. Two channels are provided. Tripping one of the channels will block erroneous rod withdrawal soon enough to prevent fuel damage. This system backs up the written sequence used by the operator for withdrawal of control rods.