Commonwealth Edison Company
LaSalle Generating Station
2601 North 21st Road
Marseilles, IL 61341-9757
Tel 815-357-6761

ComEd

April 9, 1996

United States Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, D.C. 20555

Subject: LaSalle County Nuclear Power Station Units 1 and 2
Application for Amendment of Facility Operating Licenses
NPF-11 and NPF-18, Appendix A, Technical Specifications,
Regarding Elimination of Main Steam Line High Radiation
Monitor Scram and Isolation Functions,
NRC Docket Nos. 50-373 and 50-374.

Pursuant to 10 CFR 50.90, ComEd proposes to revise Appendix A, Technical Specifications, of Facility Operating Licenses NPF-11 and NPF-18 LaSalle County Station Units 1 and 2.

The current Tech l Specification setpoint for the Main Steam Line Radiation Monit (MSLRM) system trip functions (3 times the normal full power backgrou s based on providing adequate operational margin above the normal full power background radiation level to avoid spurious isolation of the Main Steam Isolation Valves (MSIVs) and automatic reactor scrams. LaSalle County Station (LaSalle) is presently implementing the addition of a Hydrogen Water Chemistry (HWC) system to both Units. During moderate levels of hydrogen injection, with reactor power greater than 20% of rated power, main steam line radiation levels are expected to increase up to a factor of eight higher, primarily due to N-16 carryover. This submittal provides the justification for deleting scram and isolations on main steam line high radiation and inoperable trips, as well as, raising the MSLRM system alarm setpoints above the new HWC background to include the effects of increased radiation background level during hydrogen injection at full reactor power. The setpoint change and deletion of inoperable trips will not affect the current Technical Specifications.

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9. Attachment I is the Evaluation of Main Steam Line Radiation Monitor Dose Rate for LaSalle Design Basis Control Rod Drop Accident.

10. Attachment J is LaSalle Administrative procedure, LAP 100-33, Procedure - Fuel Integrity Monitoring Plan.

11. Attachment K is LaSalle Operating procedure, LOA-NB-08, Procedure - Fuel Element Failure.

This proposed amendment has been reviewed and approved by ComEd On-Site and Off-Site Review in accordance with ComEd procedures.

The elimination of the MSIV closure function and scram function associated with the MSLRM system is scheduled to be implemented during the fourth quarter of 1996 during the startup of Hydrogen Water Chemistry (HWC). The MSLRM setpoint adjustment will also be implemented during the same period. Therefore, ComEd requests that this amendment be approved by the NRC by approximately September of 1996, with an implementation time of 90 days.

ComEd is notifying the State of Illinois of this application for amendment by transmitting a copy of this letter and its attachments to the designated state official.

If there are any further questions or comments concerning this letter, please refer them to JoEllen Burns, Regulatory Assurance Supervisor, at (815) 357-6761, extension 2383.

Respectfully,

F. E. Querio
Site Vice President
LaSalle County Station

Enclosures

cc: H. J. Miller, NRC Region III Administrator
P. G. Brochman, NRC Senior Resident Inspector - LaSalle
D. M. Skay, Project Manager - NRR- LaSalle
F. Niziolek, Office of Nuclear Facility Safety - IDNS
Central File

STATE OF ILLINOIS

COUNTY OF LASALLE

Docket Nos. 50-373
50-374

IN THE MATTER OF

COMMONWEALTH EDISON COMPANY

LASALLE COUNTY - UNITS 1 & 2

AFFIDAVIT

I affirm that the content of this transmittal is true and correct to the best of my knowledge, information and belief.

for

R. E. Querio
Site Vice President
LaSalle County Station

Subscribed and sworn to before me, a Notary Public in and for the State and County above named, this 9th day of April, 1996. My Commission expires on January 24, 1997.

Lynn Ellen Gassman

Notary Public

OFFICIAL SEAL
LYNN ELLEN GASSMAN
NOTARY PUBLIC, STATE OF ILLINOIS
MY COMMISSION EXPIRES 1-24-97

ATTACHMENT A
DESCRIPTION OF THE SAFETY ANALYSIS OF THE PROPOSED CHANGES

Description of the Proposed Change

The current Technical Specification setpoint for MSLRM system trip functions, at 3 times the normal full power background, is based on providing adequate operational margin above the normal full power background radiation level to avoid spurious isolation of the MSIVs and automatic reactor scrams. LaSalle County Station (LSCS) is presently implementing the addition of a Hydrogen Water Chemistry (HWC) system to both Units. During moderate levels of hydrogen injection with reactor power greater than 20% of rated power, main steam line radiation levels are expected to increase by up to a factor of eight primarily due to N-16 carryover. This submittal provides the justification for deleting scram and isolation on main steam line high radiation and inoperable trips, as well as, raising the MSLRM system alarm setpoints above the new HWC background to include the effects of increased radiation background level during hydrogen injection at full reactor power. The setpoint change and deletion of inoperable trips will not affect the current Technical Specifications.

This proposed Technical Specification change removes the scram function and the Groups 1 and 3 isolation valve closure functions associated with the Main Steam Line Radiation Monitoring (MSLRM) system at LSCS on high radiation. The changes are as follows:

- Remove the reactor scram function
- Remove the automatic closure of the Main Steam Isolation Valves (MSIVs)
- Remove the automatic closure of the Reactor Recirculation Water Sample Line Isolation Valves and Main Steam Line Drain Isolation Valves

Elimination of these functions will improve the availability of the Station by reducing spurious scrams and isolations caused by MSLRM system. Since LSCS is proposing to eliminate automatic reactor scram and closure of the MSIVs on high radiation or inoperable trips, the references to MSLRM trip instrumentation will be removed from the Technical Specifications. The existing alarm signals, which are not part of the current Technical Specifications, will remain functional.

The general technical bases for eliminating the MSLRM system high radiation and inoperable trip functions for initiating an automatic reactor scram and automatic closure of the Main Steam Line Isolation Valves (MSIVs) is contained in General Electric's (GE's) Topical Report NEDO-

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31400A, "Safety Evaluation for Eliminating the Boiling Water Reactor Main Steam Line Isolation Valve Closure Function and Scram Function of the Main Steam Line Radiation Monitor" (Attachment E). This GE topical report provides a safety assessment for eliminating the MSLRM system high radiation trip functions for initiating an automatic reactor scram and automatic closure of the MSIVs and demonstrates that the reactor vessel isolation function and scram function of the MSLRM system are not required to ensure compliance with the requirements of 10CFR100. NEDO-31400A was approved by the NRC staff by letter dated May 15, 1991. The NRC indicated that it would be acceptable for licensees to reference this report when submitting Technical Specification changes to eliminate MSLRM system high radiation trip functions, provided that the guidance and limitations specified in NEDO-31400A and associated NRC Safety Evaluation Report are followed. NEDO-31400A is applicable to LSCS.

The proposed Technical Specification changes associated with the elimination of MSLRM high radiation trip function for initiating automatic isolation valve closure of the Main Steam Drain Lines and Reactor Recirculation System Sample Lines were not specifically addressed by NEDO-31400A. Justification for removing the automatic closure function of these additional valves is provided as part of this submittal.

The NRC has already approved similar Technical Specification changes associated with elimination of scram, and MSIV and main steam line drain isolation valve closure functions on high radiation for Hope Creek Generating Station, Docket No. 50-354 (approved by the NRC on 8/17/92), and Limerick Generating Station, Units 1 and 2, Docket Nos. 50-352/353 (approved by the NRC on 2/16/95).

Description of the Current Operating License/Technical Specification Requirement

The MSLRM system is designed to monitor radiation levels in the vicinity of main steam lines, since high radiation emanating from the lines could be indicative of fuel failure. High radiation near the main steam lines initiates a reactor scram and automatic closure of the following primary containment isolation valves:

- Eight MSIVs per unit
- Six main steam line drain valves per unit, and
- Two reactor recirculation water sample line valves per unit.

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The high radiation isolation setting is selected high enough above normal full power background radiation levels to avoid spurious isolation, yet low enough to promptly detect a gross release of fission products from the fuel prior to exceeding the Standard Review Plan (SRP) 15.4.9 acceptance criteria.

The MSLRM system consists of four gamma-sensitive instrumentation channels which monitor the gross gamma radiation from the main steam lines. The detectors are physically located near the main steam lines, just downstream of the outboard main steam isolation valves. The detectors are geometrically arranged to detect significant increases in radiation level with any number of main steam lines in operation. Their location along the main steam lines allows the earliest practical detection of a gross fuel failure.

As mentioned before, the MSIVs will close as a result of the MSLRM system isolation signal when high radiation is detected. The main steam line drain valves are also isolated along with MSIVs when high radiation is detected by the MSLRM system. The drain lines discharge into the main condenser, as do the main steam lines. Consequently the drain lines represent a branched parallel path for activity transport from the reactor vessel steam space to the condenser.

The reactor water sample line valves close automatically along with MSIVs when the MSLRM system senses high radiation. The sample line takes water from the reactor recirculation system to a sample panel in the reactor building. The combined flow from the panel is up to 800 cc/minute and contains a significant concentration of iodine because 90% of the iodine released from the fuel in a design basis Control Rod Drop Accident (CRDA) is expected to remain in the reactor water. There are two potential sample line flow paths downstream of the sample panel. For Unit 1, the normal flow path delivers the sample line flow to radwaste, while the alternate path routes the flow directly to the condenser. For Unit 2, the sample line goes to the condenser only.

In the sample line flowpath going to radwaste, water from the sample panel is delivered to an equipment drain tank in the reactor building which is vented to the steam tunnel. Water is pumped from the drain tank to a collector tank which is located in the radwaste area and vented to the room.

Bases for the Current Requirement

The main steam line radiation detectors are provided to detect a gross failure of the fuel cladding. When the high radiation is detected, a reactor trip is initiated to reduce the continued failure of fuel cladding. At the same time the MSIVs are closed to limit the release of fission products. The trip setting is high enough above normal full power background radiation levels to prevent spurious trips, yet low enough to promptly detect gross failures in the fuel cladding. No credit is taken for operation of this trip in the CRDA design basis accident analyses. However, its functional capability at the specified trip setting is required by the Technical Specifications to enhance the overall reliability of the Reactor Protection System. The high radiation condition also signals the Primary Containment Isolation System (PCIS) to initiate containment of the released fission products by automatic closure of all MSIVs, main steam line drain valves, and reactor recirculation water sample line valves. There is also an inoperative trip which produces the same isolations and scram signal.

UFSAR subsection 15.4.9.5.2 states that the MSIVs close as a result of the MSLRM system isolation signal in a design basis CRDA. To be consistent with the requirements of the Standard Review Plan (SRP) 15.4.9, all radioactive noble gases and iodine present in the reactor coolant as a result of the CRDA are transferred to the main condenser before the MSIVs close.

Description of the Need for Amending the Technical Specification

Hydrogen injection in the feedwater will result in increased concentration of N-16 in the main steam lines. This increase will impact the normal full power background radiation level and hence current setpoint value of the MSLRM alarms actuation, but will not affect the current Technical Specifications, since the setpoint will continue to be defined as 3 times full power background. The alarm function will be retained, but the values will be higher as HWC is implemented.

The MSLRM system is designed to cause a scram and isolations on high radiation near the main steam lines. The proposed amendment will remove all references to the MSLRM instrumentation from the Technical Specifications, since the MSIV isolation function will not be required per NEDO-31400A.

As documented in NEDO-31400A, there have been a number of spurious actuations of the MSLRM system at other plants causing unnecessary automatic reactor shutdowns. This action isolates the primary heat sink, imposes a large transient on the reactor vessel, and results in safety-related system actuations. Subjecting the reactor system to unnecessary vessel isolation diminishes plant reliability. NEDO-31400A documents the conclusion that the MSLRM system high radiation trip function for initiating automatic reactor scram and closure of the MSIVs, is not required to ensure compliance with the radiation dose limitation requirements stipulated in 10CFR100, even under a design basis CRDA. It demonstrates that the potential for spurious reactor shutdowns will be reduced and plant operational flexibility will be increased since the main condenser will remain available for decay heat removal following a CRDA. In addition, NEDO-31400A demonstrates that the Offgas system provides significant holdup times for radionuclides, and that the use of this system is an acceptable method of controlling unexpected radioactive material releases.

Plant modifications associated with this proposed Technical Specification change will provide a number of operational benefits. Removal of the reactor scram function from the MSLRM system will reduce potential unnecessary reactor vessel challenges associated with spurious MSLRM actuations. Removal of the isolation function of the MSIVs from the MSLRM will eliminate inadvertent pressurization events due to MSIV closures, and retain availability of the condenser for decay heat removal following a CRDA. Finally, following the highly unlikely occurrence of a CRDA, if the MSIVs remain open, steam would flow to the condenser enabling the Offgas system to process the fission product activity.

Elimination of the MSLRM system high radiation trip functions for initiating automatic closure of main steam line drain valves and reactor water sample line valves is also being proposed because the resultant offsite doses were evaluated and found to be acceptable.

The UFSAR section 7.2.2.4.9 states that the MSIVs will close as a result of a MSLRM trip signal to contain fission products released from the fuel. However, to be consistent with SRP 15.4.9, radioactive noble gases and iodine, carried over by the steam as a result of the CRDA, are transferred to the main condenser before the MSIVs close. In order to reduce the potential for unnecessary reactor shutdowns and vessel isolations due to spurious MSLRM system actuations, the BWR Owners' Group prepared and submitted topical report NEDO-31400A to provide general technical guidelines to eliminate the scram and isolation functions associated with the MSLRM system.

Description of the Amended Technical Specification Requirement

The proposed Technical Specification changes include eliminating the MSLRM system high radiation trip functions for :

- Initiating an automatic reactor scram and automatic closure of the MSIVs.
- Initiating automatic closure of the main steam line drain valves and reactor recirculation water sample valves.

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The proposed changes associated with eliminating the MSLRM high radiation trip function for initiating automatic closure of main steam line drain valves and reactor water sample line valves were not specifically evaluated in NEDO-31400A. Consequently, a plant specific evaluation of the radiological consequences of eliminating automatic isolation of these additional lines has been performed using the same analytical assumptions.

The following is a list of specific changes included in this proposed Technical Specification request:

1. Table 2.2.1-1 (Reactor Protection System Instrumentation Setpoints) - Delete setpoint requirements for "Main Steam Line Radiation - High" (i.e., Table Item 6).

2. Bases 2.2.1 (Reactor Protection System Instrumentation Setpoints) - Delete reference to ".... high steam line radiation,...." (i,e., a section of Item 5).

3. Bases 2.2.1 (Reactor Protection System Instrumentation Setpoints) - Delete paragraph "Main Steam Line Radiation - High" (i.e., Item 6).

4. Table 3.3.1-1 (Reactor Protection System Instrumentation) - Delete operational condition requirements for "Main Steam Line Radiation - High" (i.e.. Table Item 6).

5. Table 3.3.1-1 (Reactor Protection System Instrumentation - Action) - Delete action statement reference to main steam line isolation valves (i.e., Action 5).

6. Table 3.3.1-2 (Reactor Protection System Response Times) - Delete response time requirement for "Main Steam Line Radiation - High" (i.e., Table Item 6).

7. Table 4.3.1.1-1 (Reactor Protection System Instrumentation Surveillance Requirements) - Delete surveillance requirements for "Main Steam Line Radiation - High" (i.e. Table Item 6).

8. Table 3.3.2-1 (Isolation Actuation Instrumentation) - Delete operational condition requirements for "Main Steam Line Radiation - High" (i.e., Table Item A.1.c.1).

9. Table 3.3.2-2 (Isolation Actuation Instrumentation Setpoints) - Delete setpoint requirements for "Main Steam Line Radiation - High" (i.e., Table Item A.1.c.1).

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10. Table 3.3.2-3 (Isolation System Instrumentation Response Time) - Delete response time requirements for "Main Steam Line Radiation - High" (i.e., Table Item A.1.c.1).

11. Table 3.3.2-3 (Isolation System Instrumentation Response Time) - Delete table notation "***" concerning response time testing of radiation detectors.

12. Table 4.3.2.1-1 (Isolation Actuation Instrumentation Surveillance Requirements) - Delete surveillance requirements for "Main Steam Line Radiation - High" (i.e. Table Item A.1.c.1).

Although not specifically addressed by the Technical Specifications, LSCS intends to retain the MSLRM alarm setpoints at 1.5 and 3 times the full power background during power operation to initiate both a Hi and Hi-Hi radiation alarm. Due to HWC implementation at the LSCS units, the increase in background radiation level during hydrogen injection at full power has been included in this submittal by redefining the full power radiation background for the main steam lines.

Bases for the Amended Technical Specification Request

The NRC Safety Evaluation Report (SER) in topical report NEDO-31400A concluded that the removal of automatic reactor shutdown and MSIV closure trips from the MSLRM system is acceptable because the calculated radiological release consequences of the bounding CRDA will not exceed acceptable dose limits as specified in 10CFR100 and the SRP. The report has shown that there is essentially no reasonable radiological consequence benefit in a design basis CRDA of retaining the MSLRM associated reactor scram and MSIV isolation function. Licensees are allowed to reference NEDO-31400A in support of their Technical Specification change requests, provided they meet three general conditions. Therefore, each condition is restated below followed by the LSCS response demonstrating compliance.

Condition 1

The applicant demonstrates that the assumptions with regard to input values (including power per assembly, chi/Q, and decay times) that are made in the generic analysis bound those for the plant.

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Response

This criterion requires that the plant specific assumptions for the LSCS design basis CRDA must be compared with those used in the NEDO-31400A evaluation.

The referenced NEDO report presents offsite dose analyses for each of the following two scenarios. Scenario 1 was based upon the assumptions of SRP 15.4.9, which are currently applicable to a design basis CRDA in which the MSIV closure function of the MSLRM system is operational. Scenario 2 was based upon the postulated event that MSIV closure did not occur as a result of a high radiation signal from the MSLRM system, and the offgas treatment system continued to operate after the event. The design basis CRDA fission product activity release from fuel was used in each analysis since that value was shown to be bounding for events in which the MSLRMs would be expected to scram the reactor and initiate closure of the MSIVs and other small containment isolation valves, as discussed later.

In Table 1, the assumptions used in the existing design basis CRDA analysis applicable to LSCS (References a and b) are compared with assumptions used in the NEDO-31400A analysis for Scenario 1. Both analyses are based on SRP 15.4.9. The LSCS units are, either the same as, or included within the bounds of the NEDO analysis with one exception. In the NEDO analysis, the condenser is isolated immediately following the event and then leaks to the environment at ground level at a rate of 1% per day. Thus, for an event which occurs at low power when condenser vacuum is being maintained by the Mechanical Vacuum Pump (MVP), automatic isolation of the MVP release path is credited. At LSCS, by procedure LOA-NB-08, the MVP is manually tripped by the operator no later than 15 minutes after a Hi-Hi alarm is received at 3 times full power radiation background. The LSCS design basis analysis is based on occurrence of a design basis CRDA at low power with the MVP exhausting the condenser at 2850 cfm through the plant stack elevated release point. After the MVP is tripped, the condenser leak rate is the same as that stated in the NEDO analysis (1% per day). The LSCS analysis therefore differs in this respect from the generic analysis, but analytically results in acceptable doses. The calculated 2-hour radiological consequences at the Exclusion Area Boundary (EAB) are 41.3 rem for thyroid, and 5.9 rem for whole body. These values are 13.8% and 23.6% of the respective 10CFR100 limits of 300 rem and

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25 rem, and are within the SRP 15.4.9 acceptance criteria of 75 rem and 6 rem (References a and b). This is the current licensing basis for LSCS.

Scenario 2 of NEDO-31400A considers the potential impact of removing the automatic MSIV trip. If the event occurs at low power when the Steam Jet Air Ejector (SJAE) is not operating, the consequences calculated under Scenario 1 do not change as a result of removing the trip, because under SRP 15.9.4 assumptions applied in that analysis, all the activity in the steam is transported to the main condenser before the MSIVs close. The activity available for release from the condenser and the evaluation for release of activity would not differ from Scenario 1. If the event occurs at a higher power level with the SJAE operating, some portion of the activity in the condenser would be pumped into the Offgas treatment system and would thus be provided a different release path to the environment. The fraction of condenser activity transported to the Offgas system release path in a particular scenario would depend on how long the SJAE is in operation. For the purpose of evaluating the impact of releasing activity from the condenser to the environment via the alternate release path, the NEDO Scenario 2 analysis was based on the conservative approach that all of the activity in the condenser was transferred to the Offgas system. In the same analysis it is concluded that charcoal contained in the Offgas system charcoal vessels will retain the iodine activity indefinitely and does not contribute to offsite dose. Credit is taken for decay of the noble gases prior to release from the charcoal. The NEDO analysis conservatively asserts that the noble gas hold-up times in the charcoal are based on continued operation at full flow conditions and bases the dose assessment on release of all krypton and xenon from the beds during the same 2-hour window. No credit is taken for termination of the release as a consequence of detecting high offgas activity.

Table 2 provides a comparison of assumptions for LSCS with the Scenario 2 analysis. The 2-hour whole body dose at the EAB for LSCS is calculated using the following assumptions:

- While the release path to the environment is a controlled path, it is assumed conservatively for the purpose of dose analysis that no action occurs to terminate or limit the release of activity from the charcoal.

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- The noble gas delay times in the charcoal are based on the conservative assumption that non-condensable flow rate of 80 scfm is equal to the design value for high power/high steam flow operation and remains constant after the CRDA event.

- The doses from release of krypton and xenon isotopes are each calculated using the chi/Q value applicable to the 0-2 hour post-accident time period, even though krypton charcoal delay times at design flow rate are measured in hours and xenon delay times are measured in days. Thus the 2-hour time period is conservatively taken to start when the release from the charcoal starts, not when the accident occurs.

- The doses calculated using the previous assumption are combined as if the krypton and xenon were released during the same 2-hour period.

NEDO-31400A Figures 3 and 4 provide families of curves which can be used to obtain offsite dose based on plant-specific values for chi/Q and Offgas system holdup times for krypton and xenon. Activity which enters the Offgas system must pass through massive charcoal adsorber beds prior to release to the environment. The LSCS chi/Q for EAB has been conservatively taken to be the value for a stack release applicable to the 0 - 30 minute time period. This is the time interval for maximum chi/Q because a fumigation condition is postulated to exist for the first 30 minutes of a release. The charcoal filters are normally 99.9% efficient for removal of iodine in first 2 inches of the charcoal bed. The Offgas system has approximately 76 feet of charcoal in its flowpath. A plant-specific evaluation for LSCS has been performed for assessment of dose consequences due to any iodine release from the charcoal. It has been determined that the iodine is removed by adsorption in the massive charcoal beds (Attachment F).

Credit is taken for decay of the noble gases prior to release to the environment. The noble gas hold-up times of 2.9 hours for krypton and 2.2 days for xenon are derived from the UFSAR. These correspond to the non-condensable maximum design flow rate of 80 scfm. Based on the NEDO curves and LSCS data, the LSCS whole body dose at the EAB was determined to be 4.3 rem which is 17.2% of the 10CFR100 limit of 25 rem. The thyroid dose is 0 rem because, as discussed above, no iodine is released in this scenario. The Scenario 2 analysis for LSCS therefore yields consequences which are bounded

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by the doses of 5.9 rem for whole body and 41.3 rem for thyroid as determined in the existing LSCS design basis CRDA analysis (References a and b). The calculated dose is within the SRP 15.4.9 criterion of 6 rem whole body (Attachment G).

The NEDO-31400A generic analysis is based on the fission product source term for a design basis CRDA. This source term is the bounding source term for radiological consequences of eliminating the MSLRM system scram and MSIV closure function. However, the analysis did not include automatic MSLRM system valve closure functions for the main steam drain lines and reactor recirculation water sample lines. The evaluation described here demonstrates that the potential offsite dose consequences for any additional airborne release resulting from elimination of automatic closure of these isolation valves in addition to MSIV closure will be small compared to the previously evaluated design basis CRDA consequences for LSCS.

The main steam line drain valves are currently isolated along with the MSIVs by a high radiation trip signal from the MSLRM system. The (2") drain lines provide a parallel flow path from the MSIVs into the main condenser. These valves do not involve any significant difference in fission product release pathway from that of the MSIVs. The drain lines run parallel to the main steam lines through the steam tunnel to the turbine building. The main steam lines continue on to the stop valves and the drain lines go to the main condenser. Both lines are routed through areas which are locked closed due to high radiation during operation. Therefore there is no appreciable difference in radiological risk to plant personnel whether the drain lines are open or closed on a high radiation condition near the main steam lines. The elimination of the MSLRM system scram and isolation functions does not increase the calculated consequences of a design basis CRDA, because all of the activity released into the vessel steam in a CRDA is transported instantaneously into the condenser. Therefore it is concluded that elimination of the main steam drain line isolation function is consistent with elimination of the MSIV isolation function of the MSLRM system and will have no adverse effect on the calculated radiological consequences of the CRDA.

An additional isolation function which is currently associated with MSLRM system high radiation is the closure of reactor recirculation water sample line isolation valves. As with the main steam line drain valves, deletion of this isolation function was not specifically analyzed in the topical report. Following a CRDA, radioactive iodine will be present in the water flowing through the reactor recirculation sample lines.

In the normal reactor recirculation water sample line flowpath for Unit 1, the flow is routed to the reactor building equipment drain tank which is vented to the steam tunnel. Water is pumped from the drain tank to a collector tank in the radwaste building and vented to the room. The existence of vents in these tanks present the possibility that a small portion of iodine contained in the tanks could become airborne, and be transported to the stack by the reactor building and radwaste building ventilation systems, respectively, and released to the environment. An analysis of the potential offsite consequences of airborne iodine activity vented to the atmosphere from this system was performed using conservatively chosen values for iodine water-to-air partitioning and release rates to the environment. The normal flowpath results obtained for 2-hour dose at the EAB and 30-day dose at the Low Population Zone (LPZ) are compared with the design basis CRDA results in the tabulation below (Attachment H).

	EAB (rem)		LPZ (rem)	
	Thyroid	Whole Body	Thyroid	Whole Body
From Sample Line	0.01	3.74E-05	0.0935	7.65E-05
From CRDA	41.3	5.9	4.5	0.63

Thus the dose consequences for the normal sample line flowpath are very small when compared to the CRDA values.

In the alternate reactor recirculation water sample line flowpath for Unit 1 and the only flowpath for Unit 2, the flow from the sample panel is routed directly to the condenser. The potential release paths to the environment via this path are therefore similar to the potential paths considered in the design basis CRDA analysis. Based on a conservative initial condition that the MVP is operating at the beginning of the event, the calculated alternate flowpath radiological consequences for 2-hour EAB and 30-day LPZ are tabulated below (Attachment H).

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	EAB (rem)		LPZ (rem)	
	Thyroid	Whole Body	Thyroid	Whole Body
From Sample Line	8.31E-03	3.48E-05	1.96E-02	1.04E-05
From CRDA	41.3	5.9	4.5	0.63

These values for the alternate flowpath are again small when compared to the LSCS design basis dose consequences.

Thus the calculated doses are, in each case, negligible additions to the CRDA doses. The combined thyroid and whole body doses remain within the NRC acceptance criteria of 75 rem thyroid and 6 rem whole body as stated in SRP 15.4.9.

It is concluded that the assumptions for LSCS shown in Table 1 are bounded by those of the generic analysis in NEDO-31400A, with the exception of the time for which the MVP is allowed to keep running. The LSCS analysis, (approved by the NRC in Reference b) which considers the effects of this exception, shows that the offsite doses remain within the acceptance criteria of SRP 15.4.9 . Similarly, the limiting offsite doses for LSCS Scenario 2 assessment meet the SRP criteria and are bounded by those for Scenario 1.

Condition 2

The applicant includes sufficient evidence (implemented or proposed operating procedures, or equivalent commitments) to provide reasonable assurance that increased significant levels of radioactivity in the main steam lines will be controlled expeditiously to limit both occupational doses and environmental releases.

Response

Outlined below are specific actions that LSCS will take to limit occupational doses and environmental releases when the main steam line monitors detect high radiation. The MSLRM system high radiation alarms that are presently existing will be retained. The MSLRM alarm setpoint will be revised to include higher background level that would result from HWC operation at normal reactor power level. The revised setpoint will not affect offsite dose because the CRDA background radiation level is about 50 times higher than the

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expected background for HWC, as discussed in the response to Condition 3 below. Procedures are in place to address radiation alarms at 1.5 times full power background and 3 times full power background.

The affected alarm, operation, and administrative procedures will be revised to remove the automatic isolation functions and reactor auto scram at 3 times full power background and to maintain the Hi and Hi-Hi radiation alarms in the control room. Within 15 minutes of the Hi-Hi alarm, existing procedure LOA-NB-08 (Attachment K) calls for tripping the MVP manually to ensure that offsite doses do not exceed the SRP 15.4.9 limits. When main steam line Hi-Hi radiation alarm is received, existing procedures also call for ensuring that the main condenser is sealed up after tripping the MVP to minimize environmental release from the condenser following a control rod drop accident.

Fuel element failure will normally be indicated by a higher fission product activity in the primary coolant and higher offgas radiation levels. LSCS has a fuel integrity monitoring plan in place that calls for specific action levels when fuel failures are suspected. These include increase in frequency of coolant and/or offgas sample monitoring and reduction of reactor power.

The above mitigative measures will ensure that occupational doses and environmental releases will be adequately and expeditiously controlled.

Condition 3

| The applicant standardizes the MSLRM and offgas radiation monitor alarm setpoint at 1.5 times the normal Nitrogen-16 background dose rate at the monitor locations, and commits to promptly sample the reactor coolant to determine possible contamination levels in the plant reactor coolant and the need for additional corrective actions, if
| the MSLRM or offgas radiation monitors or both exceed the alarm setpoints.

Response

In the event of a CRDA, the MSLRMs will detect high radiation levels in the vicinity of the four main steam lines and will provide signals for Hi and Hi-Hi radiation alarms in the main control room at

ATTACHMENT A
DESCRIPTION OF THE SAFETY ANALYSIS OF THE PROPOSED CHANGES

approximately 1.5 and 3.0 times normal background, respectively. A plant specific evaluation (Attachment I) has been performed to evaluate the dose rate which would be detected at the MSLRMs following a design basis CRDA. The calculated exposure rate is in excess of 1600 R/hr. The original design function of the MSLRM is to provide for an early detection and protective function in the event of gross fuel failure. The Technical Specifications currently specify the MSLRM setpoint for reactor scram and MSIV isolation at 3 times the normal "full power radiation background" which is about 1.3 R/hr. During moderate hydrogen injection in the reactor feedwater, with reactor power greater than 20% of rated power, main steam line background radiation levels are expected to increase. According to Electric Power Research Institute's (EPRI) BWR Water Chemistry Guidelines - 1993 Revision, the background radiation level at the MSLRM sensors could reach as high as eight times the normal background radiation level. There will thus be a higher "full power radiation background" during HWC. The high radiation trip setpoint is expected to be in the range of 32 R/hr. Actual values will be defined as part of HWC startup testing. Since the dose rate at the MSLRMs from the design basis CRDA (1600 R/hr) is more than 50 times higher than the proposed setpoint, it is a reasonable conclusion that the design basis CRDA can be easily detected with the higher setting. The revised setpoint will be used for alarms in the main control room only.

A minor fuel element failure will normally be indicated by increasing fission product activity in the coolant and by increasing offgas radiation levels. High offgas radiation will be detected by the offgas pretreatment radiation monitor which measures the radioactivity in the condenser offgas downstream of the steam jet air ejectors. If these monitors alarm, LSCS procedures provide guidance for determining actions to manage the effects of in-core fuel failures. The action levels are set well below Technical Specification limitations to ensure that appropriate actions are taken before the license limit is reached. Thus the revised basis for the new MSLRM alarm setpoint associated with HWC implementation will not change offsite dose. The "offgas pre-treatment radiation monitor" corresponds to the "offgas radiation monitor" in Condition 3. The offgas pre-treatment radiation monitor High radiation alarm setpoint will be 1.5 times the normal full power background rate, including Nitrogen-16.

ATTACHMENT A
DESCRIPTION OF THE SAFETY ANALYSIS OF THE PROPOSED CHANGES

When fuel failure is suspected, LSCS procedure LAP-100-33, Fuel Integrity Monitoring Plan (Attachment J), provides guidance on notifying appropriate departments and describes the actions to manage fuel failures. These include taking samples to ascertain reactor coolant and offgas chemistry conditions. When a main steam line Hi radiation alarm is received at 1.5 times full power background, LSCS alarm procedures call for load reduction and monitoring offgas radiation level to determine if Technical Specification limits will be exceeded. Upon receipt of a Hi-Hi main steam line radiation alarm at 3 times full power background, LSCS procedure LOA-NB-08 (Attachment K) requires the MVP to be tripped manually within 15 minutes and monitor the effects of fuel element failure and take appropriate actions.

Based on the analysis of the three conditions stated above, it is concluded that the proposed changes satisfy the guidance issued by NRC in the safety evaluation report which accepts the NEDO-31400A.

Section 2.1 of NRC safety evaluation of Reference b includes references to alarm procedures which require remote manual trip of the MVP as an immediate operator action for main steam line Hi-Hi radiation. In general, operator actions, other than verification of automatic functions, are not included in the alarm procedures at LSCS. The main steam line high radiation alarm procedures will be revised to delete the immediate operator action to trip the MVP and refer the operator to the station abnormal procedure LOA-NB-08, Fuel Element Failure (Attachment K), for manually tripping the MVP. LOA-NB-08 has an operator action to trip the MVP within 15 minutes, if it is in operation.

Schedule

The elimination of the MSIV closure function and scram function associated with the MSLRM system is scheduled to be implemented during the fourth quarter of 1996 during the startup of HWC. The MSLRM setpoint adjustment will also be implemented during the same period. Therefore, ComEd requests that this amendment be approved by the NRC by approximately October of 1996, with an implementation time of 90 days.

ATTACHMENT A
DESCRIPTION OF THE SAFETY ANALYSIS OF THE PROPOSED CHANGES

REFERENCES

a. Letter dated November 17, 1993 from J. L. Schrage to Dr. T. E. Murley; LaSalle County Nuclear Station Units 1 and 2 Application for Emergency License Amendment.

b. Letter dated July 26, 1994, Issuance of Amendments 101 to Facility Operating License No. NPF-11 and 85 to Facility Operating License No. NPF-18 for LaSalle County Station, Units 1 and 2, respectively. (TAC NOS. M87720 and M87721).

ATTACHMENT A
DESCRIPTION OF THE SAFETY ANALYSIS OF THE PROPOSED CHANGES

TABLE 1

COMPARISON OF ASSUMPTIONS FOR LASALLE CRDA WITH NEDO-31400A

PARAMETER	NEDO-31400A	LSCS
Number of Failed Fuel Rods	850	770
Mass Fraction of Failed Fuel Exceeding Melting Temperature	0.0077	0.0077
Ratio of Failed Rod Power to Core Avg. Fuel Rod Power (Peaking Factor)	1.5	1.5
Failed Rod Power (MWt/rod)	0.12	0.110
Decay Time after Full Power Operation Assumed for Fuel Rod Activity	0	0
Activity Release from Rods with Cladding Failure (%)		
Noble gases	10	10
Iodines	10	10
Activity Release from Melted Fuel (%)		
Noble gases	100	100
Iodines	50	50
Fraction of Released Activity Transported to Condenser (%)		
Noble gases	100	100
Iodines	10	10
Fraction of Condenser Activity Remaining Airborne (%)		
Noble gases	100	100
Iodines	10	10

ATTACHMENT A
DESCRIPTION OF THE SAFETY ANALYSIS OF THE PROPOSED CHANGES

TABLE 1 cont'd

PARAMETER	NEDO-31400A	LSCS
Leak Rate from Isolated Condenser (%/day)	1.0	1.0
Time to Isolate Mechanical Vacuum Pump Path (min)	0	15
Holdup/Decay in Turbine Building	None	None
Chi/Q (sec/m^3)	2.5E-03	8.4E-05/ 5.1E-04 (Stack/ Ground)

ATTACHMENT A
DESCRIPTION OF THE SAFETY ANALYSIS OF THE PROPOSED CHANGES

TABLE 2

COMPARISON OF ASSUMPTIONS FOR SCENARIO 2 RELEASE PATH

PARAMETER	NEDO-31400A	LSCS
Activity Release to Condenser	Same as Scenario 1	Same as Scenario 1
Chi/Q for Offgas Treatment System Release Point (sec/m^3)	3.0E-04	8.4E-05
Iodine Released to Environment via Offgas System Release Path	None	None
Holdup Time in Offgas Treatment System		
Krypton	NEDO Fig. 3	2.9 hours
Xenon	NEDO Fig. 4	2.2 days

ATTACHMENT B
PROPOSED AMENDMENTS TO THE LICENSE/TECHNICAL SPECIFICATIONS

REVISED PAGES	REVISED PAGES
NPF-11 (Unit 1)	NPF-18 (Unit 2)
2-4a	2-4
B 2-11	B 2-11
3/4 3-2	3/4 3-2
3/4 3-4	3/4 3-4
3/4 3-6	3/4 3-6
3/4 3-7	3/4 3-7
3/4 3-11	3/4 3-11
3/4 3-15	3/4 3-15
3/4 3-18	3/4 3-18
3/4 3-19	3/4 3-19
3/4 3-20	3/4 3-20

Unit 1

Technical Specification Pages

TABLE 2.2.1-1 (Continued)
REACTOR PROTECTION SYSTEM INSTRUMENTATION SETPOINTS

Deleted

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
5. Main Steam Line Isolation Valve - Closure	≤ 8% closed	≤ 12% closed
6. Main Steam Line Radiation - High	≤ 3.0 x full power background	≤ 3.6 x full power background
7. Primary Containment Pressure - High	≤ 1.69 psig	≤ 1.89 psig
8. Scram Discharge Volume Water Level - High	≤ 767' 5¼"	≤ 767' 5¼"
9. Turbine Stop Valve - Closure	≤ 5% closed	≤ 7% closed
10. Turbine Control Valve Fast Closure, Trip Oil Pressure - Low	≥ 500 psig	≥ 414 psig
11. Reactor Mode Switch Shutdown Position	NA	NA
12. Manual Scram	NA	NA
13. Control Rod Drive		
a. Charging Water Header Pressure - Low	≥ 1157 psig	≥ 1134 psig
b. Delay Timer	≤ 10 seconds	≤ 10 seconds

REACTOR PROTECTION SYSTEM INSTRUMENTATION SETPOINTS (Continued)

4. Reactor Vessel Water Level-Low

The reactor vessel water level trip setpoint was chosen far enough below the normal operating level to avoid spurious trips but high enough above the fuel to assure that there is adequate protection for the fuel and pressure limits.

5. Main Steam Line Isolation Valve-Closure

The main steam line isolation valve closure trip was provided to limit the amount of fission product release for certain postulated events. The MSIV's are closed automatically from measured parameters such as high steam flow, high steam line radiation, low reactor water level, high steam tunnel temperature and low steam line pressure. The MSIV's closure scram anticipates the pressure and flux transients which could follow MSIV closure and thereby protects reactor vessel pressure and fuel thermal/hydraulic Safety Limits.

6. Main Steam Line Radiation-High

The main steam line radiation detectors are provided to detect a gross failure of the fuel cladding. When the high radiation is detected a trip is initiated to reduce the continued failure of fuel cladding. At the same time the main steam line isolation valves are closed to limit the release of fission products. The trip setting is high enough above background radiation levels to prevent spurious trips yet low enough to promptly detect gross failures in the fuel cladding. No credit was taken for operation of this trip in the accident analyses; however, its functional capability at the specified trip setting is required by this specification to enhance the overall reliability of the Reactor Protection System.

7. Primary Containment Pressure-High

High pressure in the drywell could indicate a break in the primary pressure boundary systems. The reactor is tripped in order to minimize the possibility of fuel damage and reduce the amount of energy being added to the coolant. The trip setting was selected as low as possible without causing spurious trips.

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TABLE 3.3.1-1

REACTOR PROTECTION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	APPLICABLE OPERATIONAL CONDITIONS	MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM (a)	ACTION
1. Intermediate Range Monitors:			
a. Neutron Flux - High	2	3	1
	3, 4	2	2
	5(b)	3	3
b. Inoperative	2	3	1
	3, 4	2	2
	5	3	3
2. Average Power Range Monitor:(c)			
a. Neutron Flux - High, Setdown	2	2	1
	3	2	2
	5(b)	2	3
b. Flow Biased Simulated Thermal Power-Upscale	1	2	4
c. Fixed Neutron Flux-High	1	2	4
d. Inoperative	1, 2	2	1
	3	2	2
	5	2	3
3. Reactor Vessel Steam Dome Pressure - High	1, 2(d)	2	1
4. Reactor Vessel Water Level - Low, Level 3	1, 2	2	1
5. Main Steam Line Isolation Valve - Closure	1(e)	4	4
6. Main Steam Line Radiation - High	1, 2(d)	2	5

Deleted

ACTION

ACTION 1 - Be in at least HOT SHUTDOWN within 12 hours.

ACTION 2 - Verify all insertable control rods to be inserted in the core and lock the reactor mode switch in the Shutdown position within one hour.

ACTION 3 - Suspend all operations involving CORE ALTERATIONS* and insert all insertable control rods within one hour.

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ACTION 4 - Be in at least STARTUP within 6 hours.

ACTION 5 - Be in STARTUP with the main steam line isolation valves closed within 6 hours or in at least HOT SHUTDOWN within 12 hours.

ACTION 6 - Initiate a reduction in THERMAL POWER within 15 minutes and reduce turbine first stage pressure to $\leq$ 140 psig, equivalent to THERMAL POWER less than 30% of RATED THERMAL POWER, within 2 hours.

ACTION 7 - Verify all insertable control rods to be inserted within 1 hour.

ACTION 8 - Lock the reactor mode switch in the Shutdown position within 1 hour.

ACTION 9 - Suspend all operations involving CORE ALTERATIONS,* and insert all insertable control rods and lock the reactor mode switch in the SHUTDOWN position within 1 hour.

*Except movement of IRM, SRM or special movable detectors, or replacement of LPRM strings provided SRM instrumentation is OPERABLE per Specification 3.9.2.

TABLE 3.3.1-2

REACTOR PROTECTION SYSTEM RESPONSE TIMES

FUNCTIONAL UNIT	RESPONSE TIME (Seconds)
1. Intermediate Range Monitors:	
a. Neutron Flux - High*	NA
b. Inoperative	NA
2. Average Power Range Monitor*	
a. Neutron Flux - High, Setdown	NA
b. Flow Biased Simulated Thermal Power-Upscale	$\leq$ 0.09**
c. Fixed Neutron Flux - High	$\leq$ 0.09
d. Inoperative	NA
3. Reactor Vessel Steam Dome Pressure - High	$\leq$ 0.55
4. Reactor Vessel Water Level - Low, Level 3	$\leq$ 1.05
5. Main Steam Line Isolation Valve - Closure	$\leq$ 0.06
6. ~~Main Steam Line Radiation - High~~	~~NA~~ Deleted
7. Primary Containment Pressure - High	NA
8. Scram Discharge Volume Water Level - High	NA
9. Turbine Stop Valve - Closure	$\leq$ 0.06
10. Turbine Control Valve Fast Closure, Trip Oil Pressure - Low	$\leq$ 0.08#
11. Reactor Mode Switch Shutdown Position	NA
12. Manual Scram	NA
13. Control Rod Drive	
a. Charging Water Header Pressure - Low	NA
b. Delay Timer	NA

*Neutron detectors are exempt from response time testing. Response time shall be measured from the detector output or from the input of the first electronic component in the channel.

**Not including simulated thermal power time constant.

#Measured from start of turbine control valve fast closure.

TABLE 4.3.1.1-1

REACTOR PROTECTION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION[(a)]	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED
1. Intermediate Range Monitors				
a. Neutron Flux - High	S/U[(b)], S	S/U[(c)], W	R	2*
	S	W	R	3*, 4, 5
b. Inoperative	NA	W	NA	2*, 3*, 4, 5
2. Average Power Range Monitor:[(f)]				
a. Neutron Flux - High, Setdown	S/U[(b)], S	S/U[(c)], W	SA	2*
	S	W	SA	3*, 5
b. Flow Biased Simulated Thermal Power-Upscale	S, D[(g)]	S/U[(c)], Q	W[(d)(e)], SA, R[(h)]	1
c. Fixed Neutron Flux - High	S	S/U[(c)], Q	W[(d)], SA	1
d. Inoperative	NA	Q	NA	1, 2, 3, 5
3. Reactor Vessel Steam Dome Pressure - High	NA	Q	Q	1, 2
4. Reactor Vessel Water Level - Low, Level 3	S	Q	R	1, 2
5. Main Steam Line Isolation Valve - Closure	NA	Q	R	1
6. Main Steam Line Radiation - High	S	Q	R	1, 2
7. Primary Containment Pressure - High	NA	Q	Q	1, 2

DELETED

TABLE 3.3.2-1

ISOLATION ACTUATION INSTRUMENTATION

TRIP FUNCTION	VALVE GROUPS OPERATED BY SIGNAL	MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM (b)	APPLICABLE OPERATIONAL CONDITION	ACTION
A. AUTOMATIC INITIATION				
1. PRIMARY CONTAINMENT ISOLATION				
a. Reactor Vessel Water Level				
(1) Low, Level 3	7	2	1, 2, 3	20
(2) Low Low, Level 2	2, 3	2	1, 2, 3	20
(3) Low Low Low, Level 1	1, 10	2	1, 2, 3	20
b. Drywell Pressure - High	2, 7, 10	2	1, 2, 3	20
c. Main Steam Line				
1) Radiation - High	1	2	1, 2, 3	21
	3	2	1, 2, 3	22
2) Pressure - Low	1	2	1	23
3) Flow - High	1	2/line(d)	1, 2, 3	21
d. Main Steam Line Tunnel Temperature - High	1	2	1(i)(j), 2(i)(j), 3(i)(j)	21
e. Main Steam Line Tunnel ΔTemperature - High	1	2	1(i)(j), 2(i)(j), 3(i)(j)	21
f. Condenser Vacuum - Low	1	2	1, 2*, 3*	21
2. SECONDARY CONTAINMENT ISOLATION				
a. Reactor Building Vent Exhaust Plenum Radiation - High	4(c)(e)	2	1, 2, 3 and **	24
b. Drywell Pressure - High	4(c)(e)	2	1, 2, 3	24
c. Reactor Vessel Water Level - Low Low, Level 2	4(c)(e)	2	1, 2, 3, and #	24
d. Fuel Pool Vent Exhaust Radiation - High	4(c)(e)	2	1, 2, 3, and **	24

DELETED

TABLE 3.3.2-2

ISOLATION ACTUATION INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
A. AUTOMATIC INITIATION		
1. PRIMARY CONTAINMENT ISOLATION		
a. Reactor Vessel Water Level		
1) Low, Level 3	≥ 12.5 inches*	≥ 11.0 inches*
2) Low Low, Level 2	≥ -50 inches*	≥ -57 inches*
3) Low Low Low, Level 1	≥ -129 inches*	≥ -136 inches*
b. Drywell Pressure - High	≤ 1.69 psig	≤ 1.89 psig
c. Main Steam Line		
1) Radiation - High	≤ 3.0 x full power background	≤ 3.6 x full background
2) Pressure - Low	≥ 854 psig	≥ 834 psig
3) Flow - High	≤ 111 psid	≤ 116 psid
d. Main Steam Line Tunnel Temperature - High	≤ 140°F	≤ 146°F
e. Main Steam Line Tunnel Δ Temperature - High	≤ 36°F	≤ 42°F
f. Condenser Vacuum - Low	≥ 7 inches Hg vacuum	≥ 5.5 inches Hg vacuum
2. SECONDARY CONTAINMENT ISOLATION		
a. Reactor Building Vent Exhaust Plenum Radiation - High	≤ 10 mr/hr	≤ 15 mr/hr
b. Drywell Pressure - High	≤ 1.69 psig	≤ 1.89 psig
c. Reactor Vessel Water Level - Low Low, Level 2	≥ -50 inches*	≥ -57 inches*
d. Fuel Pool Vent Exhaust Radiation - High	≤ 10 mr/hr	≤ 15 mr/hr
3. REACTOR WATER CLEANUP SYSTEM ISOLATION		
a. Δ Flow - High	≤ 70 gpm	≤ 87.5 gpm
b. Heat Exchanger Area Temperature - High	≤ 181°F	≤ 187°F
c. Heat Exchanger Area Ventilation ΔT - High	≤ 85°F	≤ 91°F
d. SLCS Initiation	NA	NA
e. Reactor Vessel Water Level - Low Low, Level 2	≥ -50 inches*	≥ -57 inches*

DELETED

TABLE 3.3.2-3

ISOLATION SYSTEM INSTRUMENTATION RESPONSE TIME

TRIP FUNCTION	RESPONSE TIME (Seconds)#
A. AUTOMATIC INITIATION	
1. PRIMARY CONTAINMENT ISOLATION	
a. Reactor Vessel Water Level	
1) Low, Level 3	N/A
2) Low Low, Level 2	N/A
3) Low Low Low, Level 1	≤ 1.0*
b. Drywell Pressure - High	N/A
c. Main Steam Line	
1) Radiation - High(**)	≤ 1.0* DELETED
2) Pressure - Low	≤ 2.0*
3) Flow - High	≤ 0.5*
d. Main Steam Line Tunnel Temperature - High	N/A
e. Condenser Vacuum - Low	N/A
f. Main Steam Line Tunnel ΔTemperature - High	N/A
2. SECONDARY CONTAINMENT ISOLATION	N/A
a. Reactor Building Vent Exhaust Plenum Radiation - High	
b. Drywell Pressure - High	
c. Reactor Vessel Water Level - Low, Level 2	
d. Fuel Pool Vent Exhaust Radiation - High	
3. REACTOR WATER CLEANUP SYSTEM ISOLATION	N/A
a. ΔFlow - High	
b. Heat Exchanger Area Temperature - High	
c. Heat Exchanger Area Ventilation ΔT-High	
d. SLCS Initiation	
e. Reactor Vessel Water Level - Low Low, Level 2	
4. REACTOR CORE ISOLATION COOLING SYSTEM ISOLATION	N/A
a. RCIC Steam Line Flow - High	
b. RCIC Steam Supply Pressure - Low	
c. RCIC Turbine Exhaust Diaphragm Pressure - High	
d. RCIC Equipment Room Temperature - High	
e. RCIC Steam Line Tunnel Temperature - High	
f. RCIC Steam Line Tunnel ΔTemperature - High	
g. Drywell Pressure - High	
h. RCIC Equipment Room ΔTemperature - High	
5. RHR SYSTEM STEAM CONDENSING MODE ISOLATION	N/A
a. RHR Equipment Area ΔTemperature - High	
b. RHR Area Cooler Temperature - High	
c. RHR Heat Exchanger Steam Supply Flow High	

ISOLATION SYSTEM INSTRUMENTATION RESPONSE TIME

TRIP FUNCTION	RESPONSE TIME (Seconds)#
6. RHR SYSTEM SHUTDOWN COOLING MODE ISOLATION	N/A
a. Reactor Vessel Water Level - Low, Level 3	
b. Reactor Vessel (RHR Cut-In Permissive) Pressure - High	
c. RHR Pump Suction Flow - High	
d. RHR Area Cooler Temperature High	
e. RHR Equipment Area ΔT High	
B. MANUAL INITIATION	N/A
1. Inboard Valves	
2. Outboard Valves	
3. Inboard Valves	
4. Outboard Valves	
5. Inboard Valves	
6. Outboard Valves	
7. Outboard Valve	

TABLE NOTATIONS

* Isolation system instrumentation response time for MSIVs only. No diesel generator delays assumed.

** Radiation detectors are exempt from response time testing. Response time shall be measured from detector output or the input of the first electronic component in the channel.

\# Isolation system instrumentation response time specified for the Trip Function actuating the MSIVs shall be added to MSIV isolation time to obtain ISOLATION SYSTEM RESPONSE TIME for each valve.

DELETED

N/A Not Applicable.

TABLE 4.3.2.1-1

ISOLATION ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED
A. AUTOMATIC INITIATION				
1. PRIMARY CONTAINMENT ISOLATION				
a. Reactor Vessel Water Level				
1) Low, Level 3	S	Q	R	1, 2, 3
2) Low Low, Level 2	NA	Q	R	1, 2, 3
3) Low Low Low, Level 1	S	Q	R	1, 2, 3
b. Drywell Pressure - High	NA	Q	Q	1, 2, 3
c. Main Steam Line				
1) Radiation - High	S	Q	R	1, 2, 3 DELETED
2) Pressure - Low	NA	Q	Q	1
3) Flow - High	NA	Q	R	1, 2, 3
d. Main Steam Line Tunnel Temperature - High	NA	Q	R	1, 2, 3
e. Condenser Vacuum - Low	NA	Q	Q	1, 2*, 3*
f. Main Steam Line Tunnel Δ Temperature - High	NA	Q	R	1, 2, 3
2. SECONDARY CONTAINMENT ISOLATION				
a. Reactor Building Vent Exhaust Plenum Radiation - High	S	Q	R	1, 2, 3, and **
b. Drywell Pressure - High	NA	Q	Q	1, 2, 3
c. Reactor Vessel Water Level - Low Low, Level 2	NA	Q	R	1, 2, 3, and #
d. Fuel Pool Vent Exhaust Radiation - High	S	Q	R	1, 2, 3, and **
3. REACTOR WATER CLEANUP SYSTEM ISOLATION				
a. Δ Flow - High	S	Q	R	1, 2, 3
b. Heat Exchanger Area Temperature - High	NA	Q	Q	1, 2, 3
c. Heat Exchanger Area Ventilation ΔT - High	NA	Q	Q	1, 2, 3
d. SLCS Initiation	NA	R	NA	1, 2, 3
e. Reactor Vessel Water Level - Low Low, Level 2	NA	Q	R	1, 2, 3

Unit 2

Technical Specification Pages

TABLE 2.2.1-1

REACTOR PROTECTION SYSTEM INSTRUMENTATION SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
1. Intermediate Range Monitor, Neutron Flux-High	≤ 120 divisions of full scale	≤ 122 divisions of full scale
2. Average Power Range Monitor:		
a. Neutron Flux-High, Setdown	≤ 15% of RATED THERMAL POWER	≤ 20% of RATED THERMAL POWER
b. Flow Biased Simulated Thermal Power - Upscale		
1) Two Recirculation Loop Operation		
a) Flow Biased	≤ 0.58W + 59% with a maximum of	≤ 0.58W + 62% with a maximum of
b) High Flow Clamped	≤ 113.5% of RATED THERMAL POWER	≤ 115.5% of RATED THERMAL POWER
2) Single Recirculation Loop Operation		
a) Flow Biased	≤ 0.58W + 54.3% with a maximum of	≤ 0.58W + 57.3% with a maximum of
b) High Flow Clamped	≤ 113.5% of RATED THERMAL POWER	≤ 115.5% of RATED THERMAL POWER
c. Fixed Neutron Flux-High	≤ 118% of RATED THERMAL POWER	≤ 120% of RATED THERMAL POWER
3. Reactor Vessel Steam Dome Pressure - High	≤ 1043 psig	≤ 1063 psig
4. Reactor Vessel Water Level - Low, Level 3	≥ 12.5 inches above instrument zero*	≥ 11 inches above instrument zero*
5. Main Steam Line Isolation Valve - Closure	≤ 8% closed	≤ 12% closed
6. Main Steam Line Radiation - High	≤ 3 x full power background	≤ 3.6 x full power background
7. Primary Containment Pressure - High	≤ 1.69 psig	≤ 1.89 psig
8. Scram Discharge Volume Water Level - High	≤ 767' 5¼"	≤ 767' 5¼"
9. Turbine Stop Valve - Closure	≤ 5% closed	≤ 7% closed

Deleted

*See Bases Figure B 3/4 3-1.

REACTOR PROTECTION SYSTEM INSTRUMENTATION SETPOINTS (Continued)

4. Reactor Vessel Water Level-Low

The reactor vessel water level trip setpoint was chosen far enough below the normal operating level to avoid spurious trips but high enough above the fuel to assure that there is adequate protection for the fuel and pressure limits.

5. Main Steam Line Isolation Valve-Closure

The main steam line isolation valve closure trip was provided to limit the amount of fission product release for certain postulated events. The MSIV's are closed automatically from measured parameters such as high steam flow, high steam line radiation, low reactor water level, high steam tunnel temperature and low steam line pressure. The MSIV's closure scram anticipates the pressure and flux transients which could follow MSIV closure and thereby protects reactor vessel pressure and fuel thermal/hydraulic Safety Limits.

6. Main Steam Line Radiation-High

The main steam line radiation detectors are provided to detect a gross failure of the fuel cladding. When the high radiation is detected a trip is initiated to reduce the continued failure of fuel cladding. At the same time the main steam line isolation valves are closed to limit the release of fission products. The trip setting is high enough above background radiation levels to prevent spurious trips yet low enough to promptly detect gross failures in the fuel cladding. No credit was taken for operation of this trip in the accident analyses; however, its functional capability at the specified trip setting is required by this specification to enhance the overall reliability of the Reactor Protection System.

7. Primary Containment Pressure-High

High pressure in the drywell could indicate a break in the primary pressure boundary systems. The reactor is tripped in order to minimize the possibility of fuel damage and reduce the amount of energy being added to the coolant. The trip setting was selected as low as possible without causing spurious trips.

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TABLE 3.3.1-1

REACTOR PROTECTION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	APPLICABLE OPERATIONAL CONDITIONS	MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM (a)	ACTION
1. Intermediate Range Monitors:			
a. Neutron Flux - High	2	3	1
	3, 4	2	2
	$5^{(b)}$	3	3
b. Inoperative	2	3	1
	3, 4	2	2
	5	3	3
2. Average Power Range Monitor:(c)			
a. Neutron Flux - High, Setdown	2	2	1
	3	2	2
	$5^{(b)}$	2	3
b. Flow Biased Simulated Thermal Power-Upscale	1	2	4
c. Fixed Neutron Flux-High	1	2	4
d. Inoperative	1, 2	2	1
	3	2	2
	5	2	3
3. Reactor Vessel Steam Dome Pressure - High	$1, 2^{(d)}$	2	1
4. Reactor Vessel Water Level - Low, Level 3	1, 2	2	1
5. Main Steam Line Isolation Valve - Closure	$1^{(e)}$	4	4
6. Main Steam Line Radiation - High	$1, 2^{(d)}$	2	5

Deleted

TABLE 3.3.1-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION

ACTION STATEMENTS

ACTION 1 - Be in at least HOT SHUTDOWN within 12 hours.

ACTION 2 - Verify all insertable control rods to be inserted in the core and lock the reactor mode switch in the Shutdown position within 1 hour.

ACTION 3 - Suspend all operations involving CORE ALTERATIONS* and insert all insertable control rods within one hour.

Deleted

ACTION 4 - Be in at least STARTUP within 6 hours.

ACTION 5 - Be in STARTUP with the main steam line isolation valves closed within 6 hours or in at least HOT SHUTDOWN within 12 hours.

ACTION 6 - Initiate a reduction in THERMAL POWER within 15 minutes and reduce turbine first stage pressure to $\leq$ 140 psig, equivalent to THERMAL POWER less than 30% of RATED THERMAL POWER, within 2 hours.

ACTION 7 - Verify all insertable control rods to be inserted within 1 hour.

ACTION 8 - Lock the reactor mode switch in the Shutdown position within 1 hour.

ACTION 9 - Suspend all operations involving CORE ALTERATIONS,* and insert all insertable control rods and lock the reactor mode switch in the SHUTDOWN position within 1 hour.

*Except movement of IRM, SRM, or special movable detectors, or replacement of LPRM strings provided SRM instrumentation is OPERABLE per Specification 3.9.2.

TABLE 3.3.1-2
REACTOR PROTECTION SYSTEM RESPONSE TIMES

TABLE 3.3.1-2

REACTOR PROTECTION SYSTEM RESPONSE TIMES

FUNCTIONAL UNIT	RESPONSE TIME (Seconds)	
1. Intermediate Range Monitors:		
a. Neutron Flux - High*	NA	
b. Inoperative	NA	
2. Average Power Range Monitor*		
a. Neutron Flux - High, Setdown	NA	
b. Flow Biased Simulated Thermal Power-Upscale	≤ 0.09**	
c. Fixed Neutron Flux - High	≤ 0.09	
d. Inoperative	NA	
3. Reactor Vessel Steam Dome Pressure - High	≤ 0.55	
4. Reactor Vessel Water Level - Low, Level 3	≤ 1.05	
5. Main Steam Line Isolation Valve - Closure	≤ 0.06	
6. Main Steam Line Radiation - High	NA	Deleted
7. Primary Containment Pressure - High	NA	
8. Scram Discharge Volume Water Level - High	NA	
9. Turbine Stop Valve - Closure	≤ 0.06	
10. Turbine Control Valve Fast Closure, Trip Oil Pressure - Low	≤ 0.08#	
11. Reactor Mode Switch Shutdown Position	NA	
12. Manual Scram	NA	
13. Control Rod Drive		
a. Charging Water Header Pressure - Low	NA	
b. Delay Timer	NA	

*Neutron detectors are exempt from response time testing. Response time shall be measured from the detector output or from the input of the first electronic component in the channel.

**Not including simulated thermal power time constant.

#Measured from start of turbine control valve fast closure.

TABLE 4.3.1.1-1

REACTOR PROTECTION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION(a)	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED
1. Intermediate Range Monitors				
a. Neutron Flux - High	S/U(b),S	S/U(c), W	R	2*
	S	W	R	3*, 4, 5
b. Inoperative	NA	W	NA	2*, 3*, 4, 5
2. Average Power Range Monitor:(f)				
a. Neutron Flux - High, Setdown	S/U(b),S	S/U(c), W	SA	2*
	S	W	SA	3*, 5
b. Flow Biased Simulated Thermal Power-Upscale	S, D(g)	S/U(c), Q	W(d)(e), SA, R(h)	1
c. Fixed Neutron Flux - High	S	S/U(c), Q	W(d), SA	1
d. Inoperative	NA	Q	NA	1, 2, 3, 5
3. Reactor Vessel Steam Dome Pressure - High	NA	Q	Q	1, 2
4. Reactor Vessel Water Level - Low, Level 3	S	Q	R	1, 2
5. Main Steam Line Isolation Valve - Closure	NA	Q	R	1
6. Main Steam Line Radiation - High	S	Q	R	1, 2
7. Primary Containment Pressure - High	NA	Q	Q	1, 2

DELETED

TABLE 3.3.2-1

ISOLATION ACTUATION INSTRUMENTATION

TRIP FUNCTION	VALVE GROUPS OPERATED BY SIGNAL	MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM (b)	APPLICABLE OPERATIONAL CONDITION	ACTION
A. AUTOMATIC INITIATION				
1. PRIMARY CONTAINMENT ISOLATION				
a. Reactor Vessel Water Level				
(1) Low, Level 3	7	2	1, 2, 3	20
(2) Low Low, Level 2	2, 3	2	1, 2, 3	20
(3) Low Low Low, Level 1	1, 10	2	1, 2, 3	20
b. Drywell Pressure - High	2, 7, 10	2	1, 2, 3	20
c. Main Steam Line				
1) Radiation - High	1	2	1, 2, 3	21
	3	2	1, 2, 3	22
2) Pressure - Low	1	2	1	23
3) Flow - High	1	2/line(d)	1, 2, 3	21
d. Main Steam Line Tunnel Temperature - High	1	2	1(i)(j), 2(i)(j), 3(i)(j)	21
e. Main Steam Line Tunnel ΔTemperature - High	1	2	1(i)(j), 2(i)(j), 3(i)(j)	21
f. Condenser Vacuum - Low	1	2	1, 2*, 3*	21
2. SECONDARY CONTAINMENT ISOLATION				
a. Reactor Building Vent Exhaust Plenum Radiation - High	4(c)(e)	2	1, 2, 3 and **	24
b. Drywell Pressure - High	4(c)(e)	2	1, 2, 3	24
c. Reactor Vessel Water Level - Low Low, Level 2	4(c)(e)	2	1, 2, 3, and #	24
d. Fuel Pool Vent Exhaust Radiation - High	4(c)(e)	2	1, 2, 3, and **	24

DELETED

TABLE 3.3.2-2

ISOLATION ACTUATION INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
A. AUTOMATIC INITIATION		
1. PRIMARY CONTAINMENT ISOLATION		
a. Reactor Vessel Water Level		
1) Low, Level 3	$\geq$ 12.5 inches*	$\geq$ 11.0 inches*
2) Low Low, Level 2	$\geq$ -50 inches*	$\geq$ -57 inches*
3) Low Low Low, Level 1	$\geq$ -129 inches*	$\geq$ -136 inches*
b. Drywell Pressure - High	$\leq$ 1.69 psig	$\leq$ 1.89 psig
c. Main Steam Line		
1) Radiation - High	$\leq$ 3.0 x full power background	$\leq$ 3.6 x full background
2) Pressure - Low	$\geq$ 854 psig	$\geq$ 834 psig
3) Flow - High	$\leq$ 111 psid	$\leq$ 116 psid
d. Main Steam Line Tunnel Temperature - High	$\leq$ 140°F	< 146°F
e. Main Steam Line Tunnel Δ Temperature - High	$\leq$ 36°F	$\leq$ 42°F
f. Condenser Vacuum - Low	$\geq$ 7 inches Hg vacuum	$\geq$ 5.5 inches Hg vacuum
2. SECONDARY CONTAINMENT ISOLATION		
a. Reactor Building Vent Exhaust Plenum Radiation - High	$\leq$ 10 mr/h	$\leq$ 15 mr/h
b. Drywell Pressure - High	$\leq$ 1.69 psig	$\leq$ 1.89 psig
c. Reactor Vessel Water Level - Low Low, Level 2	$\geq$ -50 inches*	$\geq$ -57 inches*
d. Fuel Pool Vent Exhaust Radiation - High	$\leq$ 10 mr/h	$\leq$ 15 mr/h
3. REACTOR WATER CLEANUP SYSTEM ISOLATION		
a. ΔFlow - High	$\leq$ 70 gpm	$\leq$ 87.5 gpm
b. Heat Exchanger Area Temperature - High	$\leq$ 181°F	$\leq$ 187°F
c. Heat Exchanger Area Ventilation ΔT - High	< 85°	< 91°F
d. SLCS Initiation	N.A.	N.A.
e. Reactor Vessel Water Level - Low Low, Level 2	$\geq$ -50 inches*	$\geq$ -57 inches*

DELETED

TABLE 3.3.2-3

ISOLATION SYSTEM INSTRUMENTATION RESPONSE TIME

TRIP FUNCTION	RESPONSE TIME (Seconds)#
A. AUTOMATIC INITIATION	
1. PRIMARY CONTAINMENT ISOLATION	
a. Reactor Vessel Water Level	
1) Low, Level 3	N/A
2) Low Low, Level 2	N/A
3) Low Low Low, Level 1	≤ 1.0*
b. Drywell Pressure - High	N/A
c. Main Steam Line	
1) Radiation - High(**)	≤ 1.0*
2) Pressure - Low	≤ 2.0*
3) Flow - High	≤ 0.5*
d. Main Steam Line Tunnel Temperature - High	N/A
e. Condenser Vacuum - Low	N/A
f. Main Steam Line Tunnel ΔTemperature - High	N/A
2. SECONDARY CONTAINMENT ISOLATION	N/A
a. Reactor Building Vent Exhaust Plenum Radiation - High	
b. Drywell Pressure - High	
c. Reactor Vessel Water Level - Low, Level 2	
d. Fuel Pool Vent Exhaust Radiation - High	
3. REACTOR WATER CLEANUP SYSTEM ISOLATION	N/A
a. ΔFlow - High	
b. Heat Exchanger Area Temperature - High	
c. Heat Exchanger Area Ventilation ΔT-High	
d. SLCS Initiation	
e. Reactor Vessel Water Level - Low Low, Level 2	
4. REACTOR CORE ISOLATION COOLING SYSTEM ISOLATION	N/A
a. RCIC Steam Line Flow - High	
b. RCIC Steam Supply Pressure - Low	
c. RCIC Turbine Exhaust Diaphragm Pressure - High	
d. RCIC Equipment Room Temperature - High	
e. RCIC Steam Line Tunnel Temperature - High	
f. RCIC Steam Line Tunnel ΔTemperature - High	
g. Drywell Pressure - High	
h. RCIC Equipment Room ΔTemperature - High	
5. RHR SYSTEM STEAM CONDENSING MODE ISOLATION	N/A
a. RHR Equipment Area ΔTemperature - High	
b. RHR Area Cooler Temperature - High	
c. RHR Heat Exchanger Steam Supply Flow High	

DELETED

ISOLATION SYSTEM INSTRUMENTATION RESPONSE TIME

TRIP FUNCTION	RESPONSE TIME (Seconds)#
6. RHR SYSTEM SHUTDOWN COOLING MODE ISOLATION	N/A
a. Reactor Vessel Water Level - Low, Level 3	
b. Reactor Vessel (RHR Cut-In Permissive) Pressure - High	
c. RHR Pump Suction Flow - High	
d. RHR Area Cooler Temperature High	
e. RHR Equipment Area ΔT High	
B. MANUAL INITIATION	N/A
1. Inboard Valves	
2. Outboard Valves	
3. Inboard Valves	
4. Outboard Valves	
5. Inboard Valves	
6. Outboard Valves	
7. Outboard Valve	

TABLE NOTATIONS

* Isolation system instrumentation response time for MSIVs only. No diesel generator delays assumed.

** Radiation detectors are exempt from response time testing. Response time shall be measured from detector output or the input of the first electronic component in the channel.

\# Isolation system instrumentation response time specified for the Trip Function actuating the MSIVs shall be added to MSIV isolation time to obtain ISOLATION SYSTEM RESPONSE TIME for each valve.

DELETED

N/A Not Applicable.

TABLE 4.3.2.1-1

ISOLATION ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED
A. AUTOMATIC INITIATION				
1. PRIMARY CONTAINMENT ISOLATION				
a. Reactor Vessel Water Level				
1) Low, Level 3	S	Q	R	1, 2, 3
2) Low Low, Level 2	NA	Q	R	1, 2, 3
3) Low Low Low, Level 1	S	Q	R	1, 2, 3
b. Drywell Pressure - High	NA	Q	Q	1, 2, 3
c. Main Steam Line				
1) Radiation - High	S	Q	R	1, 2, 3 DELETED
2) Pressure - Low	NA	Q	Q	1
3) Flow - High	NA	Q	R	1, 2, 3
d. Main Steam Line Tunnel Temperature - High	NA	Q	R	1, 2, 3
e. Condenser Vacuum - Low	NA	Q	Q	1, 2*, 3*
f. Main Steam Line Tunnel Δ Temperature - High	NA	Q	R	1, 2, 3
2. SECONDARY CONTAINMENT ISOLATION				
a. Reactor Building Vent Exhaust Plenum Radiation - High	S	Q	R	1, 2, 3 and **
b. Drywell Pressure - High	NA	Q	Q	1, 2, 3
c. Reactor Vessel Water Level - Low Low, Level 2	NA	Q	R	1, 2, 3, and #
d. Fuel Pool Vent Exhaust Radiation - High	S	Q	R	1, 2, 3 and **
3. REACTOR WATER CLEANUP SYSTEM ISOLATION				
a. Δ Flow - High	S	Q	R	1, 2, 3
b. Heat Exchanger Area Temperature - High	NA	Q	Q	1, 2, 3
c. Heat Exchanger Area Ventilation ΔT - High	NA	Q	Q	1, 2, 3
d. SLCS Initiation	NA	R	NA	1, 2, 3
e. Reactor Vessel Water Level - Low Low, Level 2	NA	Q	R	1, 2, 3

ATTACHMENT C
SIGNIFICANT HAZARDS CONSIDERATION

This proposed Technical Specification change removes the scram function a d certain Groups 1 and 3 isolation valve closure functions associated with the Main Steam Line Radiation Monitoring (MSLRM) system at La Salle County Station (LSCS) on high radiation. The changes are as follows:

- Remove the reactor scram function
- Remove the automatic closure of the Main Steam Isolation Valves (MSIVs)
- Remove the automatic closure of the Reactor Recirculation Water Sample Line Isolation Valves and Main Steam Line Drain Isolation Valves.

Commonwealth Edison has evaluated the proposed Technical Specification Amendment and determined that it does not represent a significant hazards consideration. Based on the criteria for defining a significant hazards consideration established in 10 CFR 50.92, operation of LaSalle County Station Units 1 and 2 in accordance with the proposed amendment will not:

1) Involve a significant increase in the probability or consequences of an accident previously evaluated because:

   Redefining the full power radiation background, thus changing the MSLRM alarm setpoint, does not change the probability of occurrence of any accident which has been postulated and analyzed in the UFSAR, but will reduce the probability of the inadvertent MSIV closure transient which is an analyzed transient in the UFSAR. It does not change the probability of malfunction of any equipment important to safety associated with LOCA, Fuel Handling Accident or CRDA. It also does not change the resultant offsite radiological dose from the bounding design basis CRDA. This is based upon all radioactivity, resulting from the design basis CRDA, going to the condenser instantaneously (or independent of the actual MSLRM setpoint) in the offsite dose calculation.

   The elimination of reactor scram and isolation of MSIVs, isolation of main steam line drain valves and reactor water sample line valves, associated with the MSLRM system actuation do not introduce, mitigate, or reduce the probability of any design basis accident, or any accident, evaluated in the UFSAR. The topical report NEDO-31400A has shown that there is essentially no reasonable radiological consequence

ATTACHMENT C
SIGNIFICANT HAZARDS CONSIDERATION

benefit in a design basis CRDA of retaining the MSLRM associated reactor scram and MSIV isolation function. In addition, the probability of inadvertent scram and isolation is reduced. The proposed change will not adversely impact the operation of the RPS or PCIS with respect to performing its other intended safety functions. The proposed change will not affect the operation of other plant systems or equipment important to safety. The consequences of eliminating the automatic closure of the main steam line drain isolation valves and reactor recirculation water sample line isolation valves along with the MSIVs has been evaluated to be negligible additions to the CRDA doses. A LSCS unique analysis has demonstrated that the radiological doses as a result of design basis CRDA are acceptable.

The MSLRM system high radiation trip was intended to function in response to a CRDA which has been previously evaluated. No credit for MSIV closure was taken in the CRDA analysis since it postulates that all the radioactive material assumed to be released from the fuel is transported to the main condenser prior to MSIV closure. Furthermore, the probability of a fuel failure is independent of the operation of the MSLRM system.

By eliminating the MSLRM induced MSIV closure, the Offgas system can be utilized to reduce potential offsite doses after a CRDA. The MVP is tripped no later than 15 minutes of a Hi-Hi radiation alarm but analytically results in acceptable offsite doses.

Thus the proposed amendment will not increase the probability of any accident previously evaluated, and the elimination of the MSLRM isolation signal for MSIVs and other small containment valves will not significantly increase the consequences of a CRDA as previously evaluated.

2) Create the possibility of a new or different kind of accident from any accident previously evaluated because:

Redefining the full power radiation background, thus changing the actual MSLRM alarm setpoint, does not alter the configuration of the plant. It does not revise any logic or function of the MSLRM trip channels or add, replace, or delete any equipment important to safety. Therefore it does not introduce any new failure modes or create any possibility of a new accident which may challenge safety to the public and has not been previously analyzed. It also does not involve any equipment which either has not been evaluated previously, or may have any safety consequences to the public.

The proposed Technical Specification changes involve eliminating the MSLRM system high radiation trip function for initiating an automatic reactor scram, and automatic isolations. The proposed changes will not affect the operation of other plant systems or equipment important to safety. The MSLRM system will continue to initiate alarms as before. Plant procedures will be in place to take appropriate mitigative measures in response to a high alarm.

The isolation and reactor scram functions associated with the MSLRM system actuation was originally intended to mitigate, not prevent, a potential accident scenario such as a CRDA or gross fuel failure event. Adding or removing an electronic signal, such as the one from the MSLRM system, does not change system or hardware design within the reactor vessel pressure boundary, and therefore will not create the possibility of a new or different kind of accident from those evaluated in the UFSAR like a LOCA or CRDA during power operation. It also does not create the possibility of a new or different kind of accident outside the reactor vessel pressure boundary from those evaluated in the UFSAR, such as a LOCA or Fuel Handling Accident. Removing the isolation signal also reduces the probability of inadvertent scram and isolation.

Therefore the proposed amendment will not create the possibility of a new or different kind of accident from any accident previously analyzed.

3) Involve a significant reduction in the margin of safety because:

The current MSLRM trip Hi-Hi alarm setpoint (about 4 R/hour with full power background at 1.3 R/hour) is at 3 times the full power radiation background. As indicated in the plant unique analytical result for LSCS, the radiological reading at the MSLRMs for design basis CRDA is equivalent to over 1200 times the normal full power radiation background (1600 R/hour

ATTACHMENT C
SIGNIFICANT HAZARDS CONSIDERATION

divided by 1.3 R/hour), or 150 times the full power radiation background during peak HWC environment (since the radiation background is 8 times the normal background). Thus the safety margin was very large, and would still be quite large with the HWC background factored into the MSLRM actuation setpoint (3 x 8 x 1.3 = about 50). The Hi alarm setpoint of 1.5 times full power background likewise will have a higher safety margin. Thus there is basically no adverse consequence to the margin of safety in the basis for the LSCS technical specifications.

The proposed Technical Specification changes to eliminate the MSLRM system high radiation trip function for initiating an automatic reactor scram, and automatic closure of the MSIVs, main steam line drain isolation valves, and reactor recircuiation water sample line isolation valves do not cause radiological dose consequences to exceed the limit established by SRP 15.4.9.

Per NEDO-31400A, the elimination of MSLRM trip/scram signal will result in the reduction of potential inadvertent scrams, unnecessary safety-related actuations, undue vessel isolation, and duty challenges during normal plant operation. These can be interpreted to be a potential reduction in core damage frequency, which translates to an improvement in the margin of safety.

Thus the margin of safety as defined in the basis of the technical specifications is essentially unaffected, and is therefore acceptable.

Guidance has been provided in "Final Procedures and Standards on No Significant Hazards Considerations," Final Rule, 51 FR 7744, for the application of standards to license change requests for determination of the existence of significant hazards considerations. This document provides examples of amendments which are and are not considered likely to involve significant hazards considerations. These proposed amendments most closely fit the example of a change which may either result in some increase to the probability or consequences of a previously analyzed accident or may reduce in some way a safety margin, but where the results of the change are clearly within all acceptable criteria with respect to the system or component specified in Standard Review Plan.

ATTACHMENT C
SIGNIFICANT HAZARDS CONSIDERATION

This proposed amendment does not involve a significant relaxation of the criteria used to establish safety limits, a significant relaxation of the bases for the limiting safety system settings or a significant relaxation of the bases for the limiting conditions for operations. Therefore, based on the guidance provided in the Federal Register and the criteria established in 10 CFR 50.92(c), the proposed change does not constitute a significant hazards consideration.

ATTACHMENT D

ENVIRONMENTAL ASSESSMENT STATEMENT APPLICABILITY REVIEW

Commonwealth Edison has evaluated the proposed amendment against the criteria for identification of licensing and regulatory actions requiring environmental assessment in accordance with 10 CFR Part 51.21. It has been determined that the propoced changes meet the criteria for categorical exclusion as provided for under 10 CFR Part 51.22(c)(9). The requested changes will have no impact on the environment. This conclusion has been determined because the changes requested do not pose significant hazards considerations or do not involve a significant increase in the amounts, and no significant changes in the types of any effluent that may be released off-site. Additionally, this request does not involve a significant increase in individual or cumulative occupational radiation exposure.

ATTACHMENT E

NEDO-31400A, Safety Evaluation for Eliminating the Boiling Water Reactor Main Steam Line Isolation Valve Closure Function and Scram Function of the Main Steam Line Radiation Monitor, dated October 1992.

This proposed Technical Specification change removes the scram function and the Group 1 and 3 isolation valve closure functions associated with the MSLRM system for high radiation. The changes are as follows:

Remove the reactor scram function

Remove the automatic closure of the Main Steam Isolation Valves (MSIVs)

Remove the automatic closure of the Reactor Recirculation Water Sample Line Isolation Valves and Main Steam Line Drain Isolation Valves

Elimination of these functions will improve plant availability by reducing spurious scrams and isolations caused by MSLRM system. Since LaSalle is proposing to eliminate automatic reactor scram and closure of the MSIVs on high radiation or inoperable trips, the references to MSLRM trip instrumentation will be removed from the Technical Specifications. The existing alarm signals, which are not part of the current Technical Specifications, will remain functional.

This proposed amendment request is subdivided as follows:

1. Attachment A is a Description of the Safety Analysis of the Proposed Changes.

2. Attachment B are Proposed Changes to the Technical Specifications for Operating Licenses NPF-11 and NPF-18.

3. Attachment C is a Significant Hazards Consideration.

4. Attachment D is an Environmental Assessment Statement Applicability Review.

5. Attachment E is NEDO-31400A, Safety Evaluation for Eliminating the Boiling Water Reactor Main Steam Line Isolation Valve Closure Function and Scram Function of the Main Steam Line Radiation Monitor, dated October 1992.

6. Attachment F is a General Electric Letter, J. Chase to S. P. Brown, Regarding MSL Radiation Trip Removal.

7. Attachment G is the LaSalle MSLRM Scram and Isolation Trip Elimination - Offsite Dose Analysis for NEDO-31400 Scenario 2.

8. Attachment H is the Offsite Dose Impact of Eliminating Automatic Isolation of LaSalle Sample Lines and Main Steam Drain Lines.