



Commonwealth Edison

1400 Opus Place
Downers Grove, Illinois 60515

December 14, 1994

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, D.C. 20555

SUBJECT: LaSalle County Nuclear Power Station Units 1 and 2
Application for Amendment to Facility Operating
Licenses NPF-11 and NPF-18, Appendix A, Technical
Specifications
NRC Docket Nos. 50-373 and 50-374

Pursuant to 10 CFR 50.90, ComEd proposes to amend Appendix A, Technical Specifications, of Facility Operating Licenses NPF-11 and NPF-18 to: (A) To minimize unnecessary testing and excessively restrictive allowed outage times (AOT) for certain actuation instrumentation in the reactor protection, isolation, emergency core cooling, recirculation pump trip, reactor core isolation cooling, control rod withdrawal block, monitoring, and feedwater/main turbine trip systems for LaSalle County Station (LaSalle) Units 1 and 2. (B) Change the Feedwater/Main Turbine Trip LCO 3.3.8 action statement to achieve consistency with existing instrumentation LCOs. (C) Delete the surveillance of the APRM Neutron Flux - High, Setdown functional unit in Operational Condition 1. (D) Revise the applicability of the provisions of Specification 4.0.4 to several Reactor Protection System and Control Rod Withdrawal Block Instrumentation surveillance requirements. (E) Add the requirement to perform shiftly channel checks for applicable RPS, PCIS, ECCS, and RCIC instrumentation channels equipped with master trip units. (F) Other changes to correct typographical errors and to delete cycle specific footnotes which are no longer applicable.

The proposed amendment request is subdivided as follows:

1. Attachment A gives a description and safety analysis of the proposed changes.
2. Attachment B includes the proposed changes to the Technical Specifications pages for LaSalle Units 1 and 2.
3. Attachment C describes ComEd's evaluation performed in accordance with 10 CFR 50.92 (c), which confirms that no significant hazard consideration is involved.

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4. Attachment D provides an Environmental Assessment Applicability Review.
5. Attachment E is the Proprietary version of the General Electric Technical Specification Improvement Analysis for the Reactor Protection System for LaSalle County Station Units 1 and 2.
6. Attachment F is the Proprietary version of the General Electric Technical Specification Improvement Analysis for the Emergency Core Cooling System for LaSalle County Station Units 1 and 2.
7. Attachment G is a withholding affidavit for the General Electric Technical Specification Improvement Analysis for the Reactor Protection System.
8. Attachment H is a withholding affidavit for the General Electric Technical Specification Improvement Analysis for the Emergency Core Cooling System Actuation Instrumentation.
9. Attachment I is the Non-Proprietary version of the General Electric Technical Specification Improvement Analysis for the Reactor Protection System for LaSalle County Station Units 1 and 2.
10. Attachment J is the Non-Proprietary version of the General Electric Technical Specification Improvement Analysis for the Emergency Core Cooling System for LaSalle County Station Units 1 and 2.
11. Attachment K is a General Electric letter on Technical Specification Improvement for BWR Instrumentation.
12. Attachment L is a BWR Owners Group letter on Technical Specification Improvement Analysis for BWR Reactor Protection System.

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

It is requested that the proposal receive review and approval in a timely manner to support implementation of the amendments during the LaSalle Unit 2 refuel outage (L2R06), presently scheduled to begin in February 1995.

December 14, 1994

The attached (Attachments E and F) General Electric Technical Specification Improvement Analysis for the Reactor Protection System and General Electric Technical Specification Improvement Analysis for the Emergency Core Cooling System Actuation Instrumentation contain information proprietary to General Electric Company. In accordance with the requirements of 10CFR 2.790 (b), affidavits for these analyses are enclosed as Attachments G and H to support the withholding of these analyses for public disclosure. Also enclosed as Attachments I and J are Non-Proprietary versions of these two Analyses.

To the best of my knowledge and belief, the statements contained above are true and correct. In some respect these statements are not based on my personal knowledge, but obtained information furnished by other Commonwealth Edison employees, contractor employees, and consultants. Such information has been reviewed in accordance with company practice, and I believe it to be reliable.

Commonwealth Edison 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.

Please direct any questions you may have concerning this submittal to this office.

Very truly yours,

Gary G. Benes

Gary G. Benes
Nuclear Licensing Administrator

Subscribed and Sworn to before me
on this 14th day of
December, 1994.

Jacqueline T. Evans
Notary Public



Attachments:

- A. Description and Safety Analysis of the Proposed Changes
- B. Marked-Up Technical Specification Pages
- C. Evaluation of Significant Hazards Considerations
- D. Environmental Assessment Applicability Review
- E. Proprietary version of the General Electric Technical Specification Improvement Analysis for the Reactor Protection System
- F. Proprietary version of the General Electric Technical Specification Improvement Analysis for the Emergency Core Cooling System Actuation Instrumentation
- G. Withholding Affidavit for General Electric Technical Specification Improvement Analysis for the Reactor Protection System
- H. Withholding Affidavit for General Electric Technical Specification Improvement Analysis for the Emergency Core Cooling System Actuation Instrumentation
- I. Non-Proprietary version of the General Electric Technical Specification Improvement Analysis for the Reactor Protection System
- J. Non-Proprietary version of the General Electric Technical Specification Improvement Analysis for the Emergency Core Cooling System Actuation Instrumentation
- K. General Electric letter on Technical Specification Improvement for BWR Instrumentation
- L. BWR Owners Group letter on Technical Specification Improvement Analysis for BWR Reactor Protection System

cc: J. B. Martin, Regional Administrator - RIII
P. G. Brochman, Senior Resident Inspector - LSCS
W. D. Reckley, Project Manager - NRR
Office of Nuclear Facility Safety - IDNS

ATTACHMENT G

WITHHOLDING AFFIDAVIT FOR THE GENERAL ELECTRIC TECHNICAL SPECIFICATION
IMPROVEMENT ANALYSIS FOR THE REACTOR PROTECTION SYSTEM.

General Electric Company

AFFIDAVIT

I, **George B. Stramback**, being duly sworn, depose and state as follows:

- (1) I am Project Manager, Licensing Services, General Electric Company ("GE") and have been delegated the function of reviewing the information described in paragraph (2) which is sought to be withheld, and have been authorized to apply for its withholding.
- (2) The information sought to be withheld is contained in the GE proprietary report MDE-83-0485, Rev. 3, *Technical Specification Improvement Analysis for the Reactor Protection System for LaSalle County Station, Units 1 and 2*, Class III (GE Company Proprietary Information), dated April 1991. This information is delineated by brackets around the specific material.
- (3) In making this application for withholding of proprietary information of which it is the owner, GE relies upon the exemption from disclosure set forth in the Freedom of Information Act ("FOIA"), 5 USC Sec. 552(b)(4), and the Trade Secrets Act, 18 USC Sec. 1905, and NRC regulations 10 CFR 9.17(a)(4), 2.790(a)(4), and 2.790(d)(1) for "trade secrets and commercial or financial information obtained from a person and privileged or confidential" (Exemption 4). The material for which exemption from disclosure is here sought is all "confidential commercial information", and some portions also qualify under the narrower definition of "trade secret", within the meanings assigned to those terms for purposes of FOIA Exemption 4 in, respectively, Critical Mass Energy Project v. Nuclear Regulatory Commission, 975F2d871 (DC Cir. 1992), and Public Citizen Health Research Group v. FDA, 704F2d1280 (DC Cir. 1983).
- (4) Some examples of categories of information which fit into the definition of proprietary information are:
 - a. Information that discloses a process, method, or apparatus, including supporting data and analyses, where prevention of its use by General Electric's competitors without license from General Electric constitutes a competitive economic advantage over other companies;
 - b. Information which, if used by a competitor, would reduce his expenditure of resources or improve his competitive position in the design, manufacture, shipment, installation, assurance of quality, or licensing of a similar product;

- c. Information which reveals cost or price information, production capacities, budget levels, or commercial strategies of General Electric, its customers, or its suppliers;
- d. Information which reveals aspects of past, present, or future General Electric customer-funded development plans and programs, of potential commercial value to General Electric;
- e. Information which discloses patentable subject matter for which it may be desirable to obtain patent protection.

The information sought to be withheld is considered to be proprietary for the reasons set forth in both paragraphs (4)b. and (4)d., above.

- (5) The information sought to be withheld is being submitted to NRC in confidence. The information is of a sort customarily held in confidence by GE, and is in fact so held. The information sought to be withheld has, to the best of my knowledge and belief, consistently been held in confidence by GE, no public disclosure has been made, and it is not available in public sources. All disclosures to third parties including any required transmittals to NRC, have been made, or must be made, pursuant to regulatory provisions or proprietary agreements which provide for maintenance of the information in confidence. Its initial designation as proprietary information, and the subsequent steps taken to prevent its unauthorized disclosure, are as set forth in paragraphs (6) and (7) following.
- (6) Initial approval of proprietary treatment of a document is made by the manager of the originating component, the person most likely to be acquainted with the value and sensitivity of the information in relation to industry knowledge. Access to such documents within GE is limited on a "need to know" basis.
- (7) The procedure for approval of external release of such a document typically requires review by the staff manager, project manager, principal scientist or other equivalent authority, by the manager of the cognizant marketing function (or his delegate), and by the Legal Operation, for technical content, competitive effect, and determination of the accuracy of the proprietary designation. Disclosures outside GE are limited to regulatory bodies, customers, and potential customers, and their agents, suppliers, and licensees, and others with a legitimate need for the information, and then only in accordance with appropriate regulatory provisions or proprietary agreements.
- (8) The information identified in paragraph (2), above, is classified as proprietary because it would provide other parties, including competitors, with a valuable information regarding the application of reliability based methodology to BWR instrumentation. A substantial effort has been expended by General Electric to develop this

information in support of the BWR Owners' Group Technical Specifications Improvement Program.

- (9) Public disclosure of the information sought to be withheld is likely to cause substantial harm to GE's competitive position and foreclose or reduce the availability of profit-making opportunities. The information is part of GE's comprehensive BWR technology base, and its commercial value extends beyond the original development cost. The value of the technology base goes beyond the extensive physical database and analytical methodology and includes development of the expertise to determine and apply the appropriate evaluation process.

The research, development, engineering, and analytical costs comprise a substantial investment of time and money by GE.

The precise value of the expertise to devise an evaluation process and apply the correct analytical methodology is difficult to quantify, but it clearly is substantial.

GE's competitive advantage will be lost if its competitors are able to use the results of the GE experience to normalize or verify their own process or if they are able to claim an equivalent understanding by demonstrating that they can arrive at the same or similar conclusions.

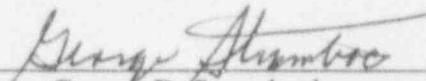
The value of this information to GE would be lost if the information were disclosed to the public. Making such information available to competitors without their having been required to undertake a similar expenditure of resources would unfairly provide competitors with a windfall, and deprive GE of the opportunity to exercise its competitive advantage to seek an adequate return on its large investment in developing these very valuable analytical tools.

STATE OF CALIFORNIA)
)
COUNTY OF SANTA CLARA) ss:

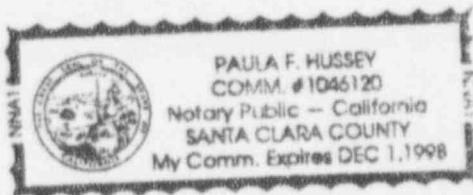
George B. Stramback, being duly sworn, deposes and says:

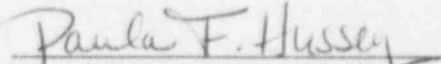
That he has read the foregoing affidavit and the matters stated therein are true and correct to the best of his knowledge, information, and belief.

Executed at San Jose, California, this 13th day of December 1994.


George B. Stramback
General Electric Company

Subscribed and sworn before me this 13th day of December 1994.




Notary Public, State of California

ATTACHMENT H

WITHHOLDING AFFIDAVIT FOR THE GENERAL ELECTRIC TECHNICAL SPECIFICATION
IMPROVEMENT ANALYSIS FOR THE EMERGENCY CORE COOLING SYSTEM ACTUATION
INSTRUMENTATION

General Electric Company

AFFIDAVIT

I, **George B. Stramback**, being duly sworn, depose and state as follows:

- (1) I am Project Manager, Licensing Services, General Electric Company ("GE") and have been delegated the function of reviewing the information described in paragraph (2) which is sought to be withheld, and have been authorized to apply for its withholding.
- (2) The information sought to be withheld is contained in the GE proprietary report RE-025, Rev. 1, *Technical Specification Improvement Analysis for the Emergency Core Cooling System Actuation Instrumentation for LaSalle County Station, Units 1 and 2*, Class III (GE Company Proprietary Information), dated April 1991. This information is delineated by brackets around the specific material.
- (3) In making this application for withholding of proprietary information of which it is the owner, GE relies upon the exemption from disclosure set forth in the Freedom of Information Act ("FOIA"), 5 USC Sec. 552(b)(4), and the Trade Secrets Act, 18 USC Sec. 1905, and NRC regulations 10 CFR 9.17(a)(4), 2.790(a)(4), and 2.790(d)(1) for "trade secrets and commercial or financial information obtained from a person and privileged or confidential" (Exemption 4). The material for which exemption from disclosure is here sought is all "confidential commercial information", and some portions also qualify under the narrower definition of "trade secret", within the meanings assigned to those terms for purposes of FOIA Exemption 4 in, respectively, Critical Mass Energy Project v. Nuclear Regulatory Commission, 975F2d871 (DC Cir. 1992), and Public Citizen Health Research Group v. FDA, 704F2d1280 (DC Cir. 1983).
- (4) Some examples of categories of information which fit into the definition of proprietary information are:
 - a. Information that discloses a process, method, or apparatus, including supporting data and analyses, where prevention of its use by General Electric's competitors without license from General Electric constitutes a competitive economic advantage over other companies;
 - b. Information which, if used by a competitor, would reduce his expenditure of resources or improve his competitive position in the design, manufacture, shipment, installation, assurance of quality, or licensing of a similar product;

- c. Information which reveals cost or price information, production capacities, budget levels, or commercial strategies of General Electric, its customers, or its suppliers;
- d. Information which reveals aspects of past, present, or future General Electric customer-funded development plans and programs, of potential commercial value to General Electric;
- e. Information which discloses patentable subject matter for which it may be desirable to obtain patent protection.

The information sought to be withheld is considered to be proprietary for the reasons set forth in both paragraphs (4)b. and (4)d., above.

- (5) The information sought to be withheld is being submitted to NRC in confidence. The information is of a sort customarily held in confidence by GE, and is in fact so held. The information sought to be withheld has, to the best of my knowledge and belief, consistently been held in confidence by GE, no public disclosure has been made, and it is not available in public sources. All disclosures to third parties including any required transmittals to NRC, have been made, or must be made, pursuant to regulatory provisions or proprietary agreements which provide for maintenance of the information in confidence. Its initial designation as proprietary information, and the subsequent steps taken to prevent its unauthorized disclosure, are as set forth in paragraphs (6) and (7) following.
- (6) Initial approval of proprietary treatment of a document is made by the manager of the originating component, the person most likely to be acquainted with the value and sensitivity of the information in relation to industry knowledge. Access to such documents within GE is limited on a "need to know" basis.
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- (8) The information identified in paragraph (2), above, is classified as proprietary because it would provide other parties, including competitors, with a valuable information regarding the application of reliability based methodology to BWR instrumentation. A substantial effort has been expended by General Electric to develop this

information in support of the BWR Owners' Group Technical Specifications Improvement Program.

- (9) Public disclosure of the information sought to be withheld is likely to cause substantial harm to GE's competitive position and foreclose or reduce the availability of profit-making opportunities. The information is part of GE's comprehensive BWR technology base, and its commercial value extends beyond the original development cost. The value of the technology base goes beyond the extensive physical database and analytical methodology and includes development of the expertise to determine and apply the appropriate evaluation process.

The research, development, engineering, and analytical costs comprise a substantial investment of time and money by GE.

The precise value of the expertise to devise an evaluation process and apply the correct analytical methodology is difficult to quantify, but it clearly is substantial.

GE's competitive advantage will be lost if its competitors are able to use the results of the GE experience to normalize or verify their own process or if they are able to claim an equivalent understanding by demonstrating that they can arrive at the same or similar conclusions.

The value of this information to GE would be lost if the information were disclosed to the public. Making such information available to competitors without their having been required to undertake a similar expenditure of resources would unfairly provide competitors with a windfall, and deprive GE of the opportunity to exercise its competitive advantage to seek an adequate return on its large investment in developing these very valuable analytical tools.

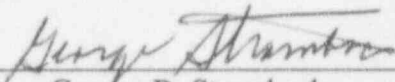
STATE OF CALIFORNIA)
)
COUNTY OF SANTA CLARA)

SS:

George B. Stramback, being duly sworn, deposes and says:

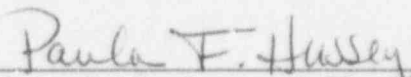
That he has read the foregoing affidavit and the matters stated therein are true and correct to the best of his knowledge, information, and belief.

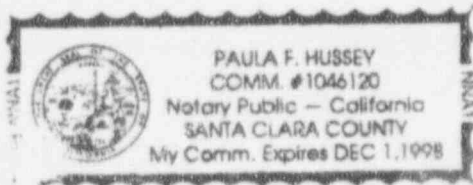
Executed at San Jose, California, this 13th day of December 1994.


George B. Stramback
General Electric Company

Subscribed and sworn before me this 13th day of December 1994.




Notary Public, State of California



ATTACHMENT A

DESCRIPTION AND SAFETY ANALYSIS OF PROPOSED CHANGES

DESCRIPTION OF THE PROPOSED CHANGE

This amendment request proposes the following:

- A. To minimize unnecessary testing and excessively restrictive allowed outage times (AOT) for certain actuation instrumentation in the reactor protection, isolation, emergency core cooling, recirculation pump trip, reactor core isolation cooling, control rod withdrawal block, monitoring, and feedwater/main turbine trip systems for LaSalle County Station (LaSalle), Units 1 and 2. This effort is part of the Technical Specification Improvement Project. The proposed changes for instrumentation are discussed and described in Appendices I through VIII.
- B. Change the Feedwater/Main Turbine Trip LCO 3.3.8 action statement to achieve consistency with existing instrumentation LCOs. This is discussed in Appendix VIII.
- C. Delete the surveillance of the APRM Neutron Flux - High, Setdown functional unit in Operational Condition 1. This is discussed in Appendix IX.
- D. Revise the applicability of the provisions of Specification 4.0.4 to several Reactor Protection System and Control Rod Withdrawal Block Instrumentation surveillance requirements. This is discussed in Appendix IX.
- E. Add the requirement to perform shiftly channel checks for applicable RPS, PCIS, ECCS, and RCIC instrumentation channels equipped with master trip units. These are discussed in Appendices I, II, III, and VI.

Other changes to correct typographical errors and to delete cycle specific footnotes which are no longer applicable are discussed in Appendix IX.

DESCRIPTION OF THE CURRENT REQUIREMENTS

- A. Technical Specification items 3/4.3.1 (Reactor Protection System Instrumentation), 3/4.3.2 (Isolation Actuation Instrumentation), 3/4.3.3 (Emergency Core Cooling System Actuation Instrumentation), 3/4.3.4 (Recirculation Pump Trip Actuation Instrumentation), 3/4.3.5 (Reactor Core Isolation Cooling System Actuation Instrumentation), 3/4.3.6 (Control Rod Withdrawal Block Instrumentation), 3/4.3.7 (Monitoring Instrumentation), and 3/4.3.8 (Feedwater/Main Turbine Trip System Actuation

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Instrumentation) contain the current requirements for the performance of channel functional tests on a weekly or monthly interval and AOT of only several hours.

- B. The Feedwater/Main Turbine Trip LCO 3.3.8 provides the actions to be taken in order to maintain the ability of the respective actuation instrumentation to initiate the feedwater system/main turbine trip system in the event of reactor vessel water level meeting or exceeding the level 8 setpoint.
- C. The APRM Neutron Flux - High, Setdown is one of many required surveillances used to demonstrate the ability of the RPS to perform its intended function.
- D. The provisions of Specification 4.0.4 specify entry to a CONDITION or OPERATIONAL CONDITION as a function of the performance of surveillance requirements associated with the applicable LCO.
- E. The performance of shiftly channel checks of the applicable RPS, PCIS, ECCS, and RCIC instrumentation channels are presently controlled solely by procedure. They are to be added to technical specifications to provide the appropriate level of control.

BASES FOR THE CURRENT TECHNICAL SPECIFICATION REQUIREMENTS

The current requirements of the above Technical Specification items provide limiting conditions for operations and surveillance requirements for the respective systems. The activities associated with the discussion of items A through E all serve to ensure that the design functions of the respective systems are preserved.

DESCRIPTION OF THE NEED FOR AMENDING THE TECHNICAL SPECIFICATIONS

- A. In late 1983 the BWR Owners' Group (BWROG) formed a Technical Specification Improvement (TSI) Committee to develop recommendations for improving the BWR Standard Technical Specifications. Commonwealth Edison Company (ComEd) is a member of the Committee. The TSI established a program for the development of reliability analyses to identify improvements for surveillance test intervals (STI) and Allowed Outage Times (AOT) for instrumentation specified in the BWR Standard Technical Specifications. The primary objective of this program was to minimize, for applicable instrumentation, unnecessary testing and excessively restrictive AOTs that could potentially degrade

ATTACHMENT A

DESCRIPTION AND SAFETY ANALYSIS OF PROPOSED CHANGES

overall plant safety and availability.

The following are examples of some of the problems experienced with the present requirements:

Scrams or engineered safety features actuations/challenges inadvertently caused during the performance of frequent surveillance tests;

AOTs which are not long enough to permit completion of surveillance tests, repairs, or maintenance on a reasonable basis;

Excessive actuation of equipment which contributes to wear-out, shortening equipment lifetimes and increasing failure rates;

Unnecessary radiation exposure to personnel performing required surveillance tests; and,

The number of required surveillance tests prevent plant personnel from performing other activities to increase the overall plant safety.

At approximately the same time, the NRC staff issued NUREG-1024, "Technical Specifications - Enhancing the Safety Impact". In the NUREG the NRC suggested that technical specification action statements be reviewed to assure that they have an adequate technical basis and do indeed minimize plant risk. This coincided well with the efforts of the BWROG. The use of reliability analyses to support engineering judgment was recognized as a primary basis for improving Technical Specification requirements. In April 1984 the TSI Committee met with the NRC staff to discuss the technical specification improvement program. At this meeting the NRC expressed agreement with the overall approach being taken by the Owners' Group. NUREG-1024 thus reinforced the BWROG's program objectives and implementation methodology.

The BWROG subsequently submitted several licensing topical reports to the NRC for review justifying STI and AOT extensions in the Technical Specifications for the Reactor Protection System (NEDC-30851P), Primary Containment Isolation System (NEDC-30851P Supplement 2, and NEDC-31677P), Emergency Core Cooling System (NEDC-30936P), and the Control Rod Block System (NEDC-30851P Supplement 1) instrumentation. The NRC evaluated and subsequently approved each of these licensing topical reports.

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DESCRIPTION AND SAFETY ANALYSIS OF PROPOSED CHANGES

In their safety evaluation reports (SER), the NRC required the applicants to:

1. Confirm the applicability of the generic analyses to their specific plant;
2. Demonstrate that instrumentation drift characteristics are bounded by the assumptions used in the General Electric (GE) analysis; and
3. Confirm that differences between the plant-specific and generic analyses were included in the plant-specific analysis.

GE confirmed the applicability of the generic analysis to LaSalle as well as the differences between it and the plant-specific analysis. These are discussed for the major instrumentation groupings in Appendices I through IV.

The SERs issued for the GE Topical Reports require confirmation that instrument setpoint drift due to extended STI is properly accounted for in the setpoint calculation methodology. This is not applicable to LaSalle because it pertains to instrumentation with calibration frequencies less than or equal to the present functional test intervals. The channel calibration extension (from 31 to 92 days) in the generic analysis applies to analog trip units when the corresponding functional test is being extended from monthly to quarterly (MDE-83-0485 Revision 1, General Electric Topical Report "Technical Specification Improvement Analysis for the Reactor Protection System for LaSalle County Station, Units 1 and 2). The LaSalle instrumentation functional tests being extended to quarterly intervals do not need to have the calibration intervals extended, because the calibration intervals are 18 months each. Also, the performance and reliability of LaSalle's instrumentation subject to channel functional testing serves as justification for extending the STI.

- B. The Feedwater/Main Turbine Trip LCO 3.3.8 as is presently written creates some confusion to the plant operating staff. Depending upon the situation, one may be required to simultaneously enter ACTION statements a. and b.
- C. Technical Specifications Table 4.3.1.1-1, "Reactor Protection System Instrumentation Surveillance Requirements", requires that the surveillance for the Average Power Range Monitor (APRM) Neutron Flux - High, Setdown functional unit be performed in Operational Condition 1, 2, 3, and 5.

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DESCRIPTION AND SAFETY ANALYSIS OF PROPOSED CHANGES

Technical Specifications Table 3.3.1-1, "Reactor Protection System Instrumentation", states operational conditions 2, 3, and 5 as the APPLICABLE OPERATIONAL CONDITIONS for the Setdown functional unit.

It is appropriate to delete the surveillance of the APRM Neutron Flux - High, Setdown functional unit in Operational Condition 1. This is proposed in order to make Table 4.3.1.1-1 consistent with Table 3.3.1-1 and Bases 2.2.1, Item 2, Average Power Range Monitor.

- D. The channel functional tests for Reactor Protection System Instrumentation items 1.a., 1.b., and 2.a. of Table 4.3.1.1-1 and Control Rod Withdrawal Block Instrumentation items 2.d., 3.a., 3.b., 3.c., 3.d., 4.a., 4.b., 4.c., and 4.d. of Table 4.3.6-1 are mode switch dependent and cannot be performed in Mode 1. Specification 4.0.4 requires that surveillances as specified in the above Tables be performed prior to entry into an Operational Condition for which the surveillance is required.
- E. LaSalle presently uses analog instrumentation as opposed to static-o-ring (SOR) switches for reactor vessel level determination in the RPS, primary containment isolation (PCIS), Emergency Core Cooling (ECCS), and Reactor Core Isolation Cooling (RCIC) systems. The SOR switches previously used for reactor vessel water level applications were not equipped with an indicator which would permit the performance of a channel check. The master trip units are now equipped with indicators which permit performing instrumentation channel checks. The requirement to perform this task is not stated in the technical specifications. A requirement to perform a shiftly channel check may now be added to the technical specifications for the applicable RPS, PCIS, ECCS, and RCIC instrumentation channels equipped with master trip units.

DESCRIPTION OF THE AMENDED TECHNICAL SPECIFICATION REQUIREMENTS

- A. For selected actuation instrumentation, the STI are to be extended from weekly or monthly to quarterly, and AOT are extended as specified in the Appendices.
- B. The revised content and structure of the Feedwater/Main Turbine Trip LCO 3.3.8 serves to ensure consistency with other instrumentation LCOs presented in this proposal, and provides more concise guidance to the operating staff.

ATTACHMENT A

DESCRIPTION AND SAFETY ANALYSIS OF PROPOSED CHANGES

- C. Technical Specifications Tables 4.3.1.1-1, "Reactor Protection System Instrumentation Surveillance Requirements", will require that surveillances for the Average Power Range Monitor (APRM) Neutron Flux - High, Setdown functional unit be performed in Operational Condition 2, 3, and 5.
- D. A footnote added to the Reactor Protection System Instrumentation Surveillance Requirements items 1.a., 1.b., and 2.a. of Table 4.3.1.1-1, and Control Rod Withdrawal Block Instrumentation Surveillance Requirements items 2.d., 3.a., 3.b., 3.c., 3.d., 4.a., 4.b., 4.c., and 4.d. of Table 4.3.6-1 excludes these instruments from being subject to the provisions of Specification 4.0.4 for a period of 24 hours after entering OPERATIONAL CONDITION 2 or 3 when shutting down from OPERATIONAL CONDITION 1.
- E. Shiftly channel checks of recently installed analog reactor vessel water level instrumentation are to be included as technical specification requirements.

BASES FOR THE AMENDED TECHNICAL SPECIFICATION REQUIREMENTS

With respect to the activities discussed in item A, analyses provided by GE demonstrate that extending AOT and STI for the subject instrumentation results in fewer reactor scrams and challenges to plant safety systems, enhanced equipment lifetimes, reduced radiation exposure to plant personnel, and reduced labor requirements that more than offset the negligible reduction in reliability associated with the extensions.

The requirements of the amended Technical Specifications for activities discussed in items B through E enhance the LCOs and surveillance requirements for the respective systems. The revisions enhance assurance that the systems are able to perform their intended design function when required to do so.

SCHEDULE REQUIREMENTS

It is requested that the proposal receive review and approval in a timely manner to support implementation of the amendments during the LaSalle Unit 2 Sixth refuel outage (L2R06), presently scheduled to begin in February 1995.

ATTACHMENT A

DESCRIPTION AND SAFETY ANALYSIS OF PROPOSED CHANGES

APPENDIX I

REACTOR PROTECTION SYSTEM INSTRUMENTATION

The following are the proposed changes to Technical Specifications Section 3/4.3.1, "Reactor Protection System Instrumentation":

A. AOT

1. LCO 3.3.1, DELETE actions a and b, and replace them with the following:
 - a. With one channel required by Table 3.3.1-1 inoperable in one or more Functional Units, place the inoperable channel and/or that trip system in the tripped condition* within 12 hours.
 - b. With two or more channels required by Table 3.3.1-1 inoperable in one or more Functional Units:
 1. Within one hour, verify sufficient channels remain OPERABLE or tripped* to maintain trip capability in the Functional Unit, and
 2. Within 6 hours, place the inoperable channel(s) in one trip system and/or that trip system** in the tripped condition*, and
 3. Within 12 hours, restore the inoperable channels in the other trip system to an OPERABLE status or tripped*.
 - c. Otherwise, take the ACTION required by Table 3.3.1-1 for the Functional Unit.
2. Replace Technical Specification 3.3.1 footnotes "*" and "***" with the following:

* An inoperable channel or trip system need not be placed in the tripped condition where this would cause the Trip Function to occur. In these cases, if the inoperable channel is not restored to OPERABLE status within the required time, the ACTION required by Table 3.3.1-1 for the Functional Unit shall be taken.

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** This ACTION applies to that trip system with the most inoperable channels; if both trip systems have the same number of inoperable channels, the ACTION can be applied to either trip system.

3. Technical Specification Table 3.3.1-1, footnote (a), change the surveillance AOT from 2 (two) hours to 6 (six) hours.
- B. Channel Check - Add requirement for a shiftly channel check of Reactor Vessel Water Level Instrumentation to Functional Unit 4 of Table 4.3.1.1-1, Reactor Vessel Water Level - Low, Level 3
- C. Channel Functional Test Interval - Technical Specification Table 4.3.1.1-1
 1. Change the test interval from Weekly to Quarterly for the following functional units:
 - a. 2.b. APRM Flow Biased Simulated Thermal Power - Upscale
 - b. 2.c. APRM Fixed Neutron Flux - High
 - c. 2.d. APRM Inoperative
 2. Change the test interval from Monthly to Quarterly for the following functional units:
 - a. 3. Reactor Vessel Steam Dome Pressure - High
 - b. 4. Reactor Vessel Water Level - Low, Level 3
 - c. 6. Main Steam Line Radiation - High
 - d. 7. Primary Containment Pressure - High
 - e. 8. Scram Discharge Volume Water Level - High
- D. Technical Specification Bases - Add the following to the bases to provide reference for the amendment:

Replace "for MSIV-Closure, TSV-Closure, TCV- Closure, and the Manual Scram" with "and surveillance and maintenance outage times";

Insert "and MDE-83-0485 Revision 3, "Technical Specification Improvement Analysis for the Reactor Protection System for LaSalle County Station, Units 1 and 2, April 1991"." at the end of the next to the last sentence of the first paragraph of the Bases.

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Insert "When a channel is placed in an inoperable status solely for performance of required surveillances, entry into LCO and required ACTIONS may be delayed, provided the associated function maintains KPS trip capability." at the end of the last sentence of the Bases paragraph.

Justification for Proposed Changes

On May 31, 1985 the BWROG submitted Licensing Topical Report NEDC-30851P, "Technical Specification Improvement Analyses for BWR Reactor Protection System", for NRC Review. The analyses documented in NEDC-30851P utilized fault tree modeling to estimate the impact of the proposed changes on the average RPS failure frequency.

The average RPS failure frequency is a function of the frequency of scram demands and the probability that the RPS is unavailable when demanded. The initiating events which require successful operation of the RPS for ensuring safe reactor shutdown were identified, and annual occurrence frequencies identified. Initiating events were divided into 3 groups, depending on the amount of diverse sensors associated with scram initiation for that event.

For each initiating event, a top-level failure event was identified using the success criteria described below. For each top-failure event, a fault tree was developed which modeled all of the components needed for generation and processing of the RPS signals. The common cause failure of components was also modeled. A fault tree analysis was then performed using the WAM series computer code, WAMCUT, to obtain the major failure cut sets that contribute to the top failure event probability. The failure cut sets obtained were then analyzed using the FRANTIC III computer code to determine the average RPS system unavailability upon demand.

The average RPS unavailability was calculated for each initiating event group based on inputs which included component failure rates, common cause failure rates, human error rates, testing intervals, as well as test and repair times. Sensitivity studies determined the resultant impact on the average RPS unavailability and the total RPS failure frequency. The resultant effect on the average RPS failure frequency was determined by varying the STI and AOT.

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The acceptance guideline used by the BWROG for the proposed changes is based upon a net change in risk. The net change is the difference between the increase in risk resulting from the proposed changes and the decrease in risk that would result from the reduced likelihood of inadvertent scrams. The BWROG considered the proposed changes to be acceptable if the net change in risk was determined to be insignificant. It was concluded that the slight increase in risk due to these extensions is offset by the benefits achieved from a reduced number of challenges to safety systems.

NEDC-30851P proposed to extend the repair and test AOT from 1 hour and 2 hours to 12 hours and 6 hours, respectively. Test intervals were proposed to be increased from their current weekly and monthly intervals to 3 month intervals without significantly affecting the overall RPS failure frequency.

The NRC concurred and subsequently issued a safety Evaluation Report (SER) on July 15, 1987, approving NEDC-30851P, to extend the STI and AOT for RPS instrumentation following its review and acceptance of this report. The BWROG subsequently issued the approved version of the Topical, NEDC-30851P-A.

GE addressed the applicability of the generic analyses to LaSalle, and confirmed that the differences between the LaSalle instrumentation and the generic model were included in the plant specific analysis. This is documented in General Electric Topical Report, "Technical Specification Improvement Analysis for the Reactor Protection System for LaSalle County Station, Units 1 and 2," MDE-83-0485 Revision 2. This report concluded that differences between LaSalle's RPS configuration and the generic model do not significantly affect the conclusions drawn in the generic analysis, so that the generic analysis does indeed apply to LaSalle. MDE-83-0485 Revision 3 accounts for modifications which could have affected the original plant specific analysis. The Topical concluded that the RPS system modifications do not affect the original analysis.

The analysis did not extend the STI for the reactor vessel water level instrumentation. At the time SOR switches were used in that application. These switches exhibit a relatively high degree of instrument setpoint drift, and since it was planned to replace them with a more reliable analog system, they weren't evaluated. They have all been replaced with analog instrumentation. The results of

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the analysis now extend to reactor vessel water level, per Revision 3 of the Topical.

Plant modifications installed since Revision 3 of the Topical was issued were evaluated to assess any effects upon the conclusions drawn in the generic and plant specific analyses. The modifications since installed do not affect the systems with respect to the extension of STI and AOT.

Revision 3 also stated that surveillance test procedures for LaSalle specify that the RPS scram contactors should be tested with four actuations per scram contactor pair in each trip logic channel for the channel functional tests of APRMs. This is no longer accurate since the APRM functional test procedures have been revised to minimize the amount of time that the unit is in a half-scram condition during testing. A scram contactor pair of a given RPS bus is actuated no more than two times during functional testing of the APRMs. This prevents surveillance testing from jeopardizing plant operations, and is consistent with Reference 3 of the Topical, since one actuation should be sufficient to determine component failure during the surveillance testing.

The BWROG sent a letter to B.K. Grimes (Nuclear Reactor Regulation), "BWR Owners' Group (BWROG) Topical Reports On Technical Specification Improvement Analysis For BWR Reactor Protection Systems - Use For Relay and Solid State Plants (NEDC-30884 and NEDC-30851P)", BWROG-92102, dated November 4, 1992, to address an NRC concern raised by the model Technical Specification Action 3.3.1a proposed in the generic analysis of NEDC-30851P, and in the LaSalle specific analysis (General Electric letter to R. H. Mirochna (Commonwealth Edison Company), "Technical Specification Improvement For BWR Instrumentation Transmittal of Deliverables LaSalle County Station", EBO-90-246, May 1, 1991). The NRC felt that the wording of Action 3.3.1a in these analyses would allow continued plant operation for up to 12 hours with a combination of failures that could prevent a reactor scram function from completing its logic when called upon, thereby resulting in a loss of function.

The wording proposed for Action 3.3.1 of BWROG-92102 is incorporated into this submittal. This clarifies operator actions in the event of the loss of two instrument channels. A discussion of the application and justification for the revised Action is provided in Enclosure 2 of BWROG-92102. LaSalle endorses the

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updated wording of the statement which reinforces prevention of an extended loss-of-function period in an RPS Functional Unit.

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The following are the proposed changes to Technical Specifications Section 3/4.3.2, "Isolation Actuation Instrumentation":

A. AOT

1. LCO 3.3.2, DELETE actions b and c, and replace them with the following:

- b. With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip System Requirement for one trip system, either

1. Place the inoperable channel(s) and/or trip system in the tripped condition* within

- a) 1 hour for trip functions without an OPERABLE channel,
- b) 12 hours for trip functions common to RPS Instrumentation, and
- c) 24 hours for trip functions not common to RPS Instrumentation.

or

2. Take the ACTION required by Table 3.3.2-1.

- c. With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip System requirement for both trip systems,

1. Place at least one trip system** in the tripped condition*** within one hour, and

2. a) Place the inoperable channel(s) in the remaining trip system in the tripped condition*** within

- 1) 1 hour for trip functions without an OPERABLE channel,
- 2) 12 hours for trip functions common to RPS Instrumentation, and
- 3) 24 hours for trip functions not common to RPS Instrumentation,

or

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- b) Take the ACTION required by Table 3.3.2-1.
- 2. Change the AOT of Table 3.3.2-1, Footnote "(b)" from 2 (two) hours to 6 (six) hours.
- B. Channel Check - Table 4.3.2.1-1, add the requirement for a shiftly channel check of reactor vessel water level instrumentation Trip Functions A.1.a.1), Reactor Vessel Water Level - Low, Level 3; A.1.a.3), Reactor Vessel Water Level - Low Low Low, Level 1; and A.6.a, Reactor Vessel Water Level - Low, Level 3.
- C. Channel Functional Test Interval - Technical Specification Table 4.3.2.1-1: Change the channel functional test interval from Monthly to Quarterly for the following trip functions:
 - 1. Primary Containment Isolation
 - a. A.1.a.1) Reactor Vessel Water Level - Low, Level 3
 - b. A.1.a.2) Reactor Vessel Water Level - Low Low, Level 2
 - c. A.1.a.3) Reactor Vessel Water Level - Low Low Low, Level 1
 - d. A.1.b Drywell Pressure - High
 - e. A.1.c.1) Main Steam Line Radiation - High
 - f. A.1.c.2) Main Steam Line Pressure - Low
 - g. A.1.c.3) Main Steam Line Flow - High
 - h. A.1.d Main Steam Line Tunnel Temperature - High
 - i. A.1.e. Condenser Vacuum - Low
 - j. A.1.f Main Steam Line Tunnel Temperature - High
 - 2. Secondary Containment Isolation
 - a. A.2.a Reactor Building Vent Exhaust Plenum Radiation - High
 - b. A.2.b Drywell Pressure - High
 - c. A.2.c Reactor Vessel Water Level - Low Low, Level 2
 - d. A.2.d Fuel Pool Vent Exhaust Radiation - High

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3. Reactor Water Cleanup System Isolation

- a. A.3.a Δ Flow - High
- b. A.3.b. Heat Exchanger Area Temperature - High
- c. A.3.c Heat Exchanger Area Ventilation Δ T - High
- d. A.3.e Reactor Vessel Water Level - Low Low, Level 2

4. Reactor Core Isolation Cooling System Isolation

- a. A.4.a RCIC Steam Line Flow - High
- b. A.4.b RCIC Steam Supply Pressure - Low
- c. A.4.c RCIC Turbine Exhaust Diaphragm Pressure - High
- d. A.4.d RCIC Equipment Room Temperature - High
- e. A.4.e RCIC Steam Line Tunnel Temperature - High
- f. A.4.f RCIC Steam Line Tunnel Δ Temperature - High
- g. A.f.g Drywell Pressure - High
- h. A.4.h RCIC Equipment Room Δ Temperature - High

5. RHR System Steam Condensing Mode Isolation

- a. A.5.a RHR Equipment Area Δ Temperature - High
- b. A.5.b RHR Area Cooler Temperature - High
- c. A.5.c RHR Heat Exchanger Steam Supply Flow - High

6. RHR System Shutdown Cooling Mode Isolation

- a. A.6.a Reactor Vessel Water Level - Low, Level 3
- b. A.6.b Reactor Vessel (RHR Cut-in Permissive) Pressure - High
- c. A.6.c RHR Pump Suction Flow - High
- d. A.6.d RHR Area Temperature - High
- e. A.6.e RHR Equipment Area Δ T - High

D. Footnotes

- 1. In footnote * of LCO 3.3.2, change "2 hours" to "6 hours".

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- E. Technical Specification Bases - Add the following statement to the bases to provide reference for the amendment:

Specified surveillance intervals and surveillance and maintenance outage times have been determined in accordance with NEDC-30851P-A, Supplement 2, "Technical Specification Improvement Analyses for BWR Isolation Instrumentation Common to RPS and ECCS Instrumentation", March 1989, and with NEDC-31677P-A, "Technical Specification Improvement Analysis for BWR Isolation Actuation Instrumentation", July 1990. When a channel is placed in an inoperable status solely for performance of required surveillances, entry into LCO and required ACTIONS may be delayed, provided the associated function maintains primary containment isolation capability.

Justification for Proposed Changes

On August 29, 1986 the BWROG submitted Licensing Topical Report NEDC-30851P, Supplement 2, "Technical Specification Improvement Analyses for BWR Isolation Instrumentation Common to RPS and ECCS Instrumentation" for NRC review. On June 27, 1989 the BWROG submitted Licensing Topical Report NEDC-31677P, "Technical Specification Improvement Analyses for BWR Isolation Actuation Instrumentation" for NRC review. The analyses for RPS and ECCS technical specification improvements performed for the BWR Owners' Group provided the bases for increasing the STI for various RPS and ECCS instruments.

A number of these instruments perform functions which are common to other PCIS instruments. The Owners' Group provided a generic analysis in General Electric Topical Report, "Technical Specification Improvement Analyses for BWR Isolation Instrumentation Common to RPS and ECCS Instrumentation," NEDC-30851P, Supplement 2, July 1986. NEDC-30851P, Supplement 2, July 1986, "Technical Specification Improvement Analyses for BWR Isolation Instrumentation Common to RPS and ECCS Instrumentation," extends the results of Topical Reports NEDC-30851P and NEDC-30936 Parts 1 and 2, to the common isolation instrumentation to obtain the maximum benefit from the RPS and ECCS analyses. This consisted of increasing the STI from one to three months, and extending AOT for test from 2 to 6 hours, and extending AOT for repair from 1 to 12 hours.

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The generic analysis provided by NEDC-31677P, "Technical Specification Improvement Analyses for BWR Isolation Actuation Instrumentation", extended the results to isolation actuation instrumentation not common to RPS and ECCS instrumentation. This consisted of increasing the STI from one to three months, extending AOT for test from 2 to 6 hours, and extending the AOT for repair from 1 to 24 hours.

Technical Specification requirements for isolation instrumentation were originally established largely on the basis of RPS and ECCS requirements. The STI and AOT generally do not need to be more stringent for isolation than for RPS or ECCS. Even though isolation is a safety function, failure to isolate would by itself not create an accident. The analysis of Supplement 2 concluded that the impact on the average isolation unavailability for an affected isolation subfunction due to the proposed changes was determined to be negligible (less than 1%) when combined with individual valve failure probabilities. The analyses demonstrate that the individual valve failure probabilities dominate isolation failure probability.

On January 6, 1989, the NRC approved NEDC-30851P, Supplement 2, and issued an SER, allowing extensions of the STI and AOT for isolation instrumentation common to the RPS and ECCS. The BWROG subsequently issued the approved version of the Topical, NEDC-30851P-A, Supplement 2.

On June 18, 1990, the NRC approved NEDC-31677P, and issued an SER, allowing extensions of the STI and AOT for isolation instrumentation not common to the RPS and ECCS instrumentation. The BWROG subsequently issued the approved version of the Topical, NEDC-31677P-A.

The confirmation for the applicability of the generic analysis for the common RPS instrumentation at LaSalle is documented in General Electric Topical Report MDE-83-0485 Revision 3, "Technical Specification Improvement Analysis for the Reactor Protection System for LaSalle County Station, Units 1 and 2". Topical Report RE-025 Revision 3, "Technical Specification Improvement Analysis for the Emergency Core Cooling System Actuation Instrumentation for LaSalle County Station, Units 1 and 2", confirmed the applicability of the generic analysis for the common ECCS instrumentation. For isolation instrumentation which is not common, analyses were provided in NEDC-31677P that bounds the plant specific differences

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identified in General Electric letter ERO-90-246, to R. H. Mirochna (Commonwealth Edison Company), "Technical Specification Improvement For BWR Instrumentation Transmittal of Deliverables LaSalle County Station", dated May 1, 1991. Appendix C of NEDC-31677P provides the LaSalle Unit 1 and 2 surveillance and calibration intervals that were included in the study. LaSalle concludes that the generic analyses bound the LaSalle specific analyses.

Plant modifications installed since the issuance of the plant specific analyses were evaluated to assess any effects upon the conclusions drawn in the generic and plant specific analyses. The modifications installed do not affect the systems with respect to the extension of STI and AOT.

The change to LCO 3.3.2 strengthens preventing a "Loss of Function" for isolation actuation instrumentation, by clarifying operator actions to be taken in the event of the loss of two instrument channels. This modified version of the AOT wording does not alter the original meaning or intent of the GE Licensing Topical Reports or the BWR Owners' Group clarifications. Clarification of the changes to the LCO as proposed in Topical Report NEDC-31677P was provided in GE letter OG90-579-32A from W. P. Sullivan and J. F. Klapproth (GE) to M. L. Wohl (NRR), dated June 25, 1990, "Implementation Enhancements to Technical Specification Changes Given in Isolation Actuation Instrumentation Analysis".

The change is similar in nature to that proposed by Niagara Mohawk in their Technical Specification Amendment submittal for Nine Mile Point Unit 2. The change is an evolvement of the LCO proposal previously discussed in GE letter OG90-579-32A. This resulted from a discussion between Mr. D. Brinkman and Mr. C. Schulten (NRR), and Mr. W. Drews (Niagara Mohawk Power Corporation). The purpose of the discussion was to resolve the loss of function issue by refining the wording of the LCO Actions.

The Nine Mile Point Unit 2 amendment was submitted via letter from B.R. Sylvia (Niagara Mohawk Power Corporation) to US NRC dated March 4, 1993 (NMP2L 0740)); Nine Mile Point Unit 2 Technical Specification Amendment Request (Docket No. 50-410, NPF-69). On May 11, 1993, the NRC approved the submittal and issued Technical Specification amendment No. 41 to Nine Mile Point Nuclear Station, Unit 2 (TAC No. M85168).

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The change as proposed for the LaSalle LCO provides measures consistent with those of Nine Mile Point Unit 2 in preventing a loss of function.

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EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION

The following are the proposed changes to Technical Specifications Section 3/4.3.3, "Emergency Core Cooling System Actuation Instrumentation":

A. AOT - Table 3.3.3-1

1. Change the inoperable AOT of Footnote (a) from two (2) to six (6) hours.
2. Extend the AOT to 24 hours for applicable ECCS instrumentation:

Action 30 a., 31, 33, 35 a., and 38 a.: Change "one hour" to "24 hours".

Action 32: Change "ECCS inoperable." to "ECCS inoperable within 24 hours".

Action 34: Change "8 hours" to "24 hours".

B. Channel Check - Table 4.3.3.1-1: Add the requirement for a shiftly channel check of the following reactor vessel water level instrumentation trip functions:

1. A.1.a. Division 1 Trip System RHR-A (LPCI MODE) and LPCS System Reactor Vessel Water Level - Low Low Low, Level 1.
2. Division 1 Trip System Automatic Depressurization System Trip System "A":
 - a. A.2.a. Reactor Vessel Water Level - Low Low Low, Level 1.
 - b. A.2.d. Reactor Vessel Water Level - Low, Level 3.
3. B.1.a. Division 2 Trip System RHR B and C (LPCI Mode) Reactor Vessel Water Level - Low Low Low, Level 1.
4. Division 2 Trip System Automatic Depressurization System Trip System "B":

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- a. B.2.a. Reactor Vessel Water Level - Low Low Low, Level 1.
- b. B.2.d. Reactor Vessel Water Level - Low, Level 3.

5. Division 3 Trip System HPCS System:

- a. C.1.a. Reactor Vessel Water Level - Low Low, Level 2.
- b. C.1.c. Reactor Vessel Water Level - High, Level 8.

- C. Channel Functional Test Interval - Technical Specification Table 4.3.3.1-1. Change the test interval from Monthly to Quarterly for the following instrumentation channels:

Division 1 Trip System

1. RHR-A (LPCI Mode) and LPCS System

- a. A.1.a. Reactor Vessel Water Level - Low Low Low, Level 1
- b. A.1.b. Drywell Pressure - High
- c. A.1.c. LPCS Pump Discharge Flow - Low
- d. A.1.d. LPCS and LPCI A Injection Valve Injection Line

Pressure Low Interlock

- e. A.1.e. LPCS and LPCI A Injection Valve Reactor Pressure Low Interlock
- f. A.1.f. LPCI Pump A Start Time Delay Relay
- g. A.1.g. LPCI Pump A Flow - Low

2. Automatic Depressurization System Trip System "A"

- a. A.2.a. Reactor Vessel Water Level - Low Low Low, Level 1
- b. A.2.b. Drywell Pressure - High
- c. A.2.c. Initiation Timer
- d. A.2.d. Reactor Vessel Water Level - Low, Level 3
- e. A.2.e. LPCS Pump Discharge Pressure - High
- f. A.2.f. LPCI Pump A Discharge Pressure - High
- h. A.2.h. Drywell Pressure Bypass Timer

Division 2 Trip System

1. RHR B and C (LPCI Mode)

- a. B.1.a. Reactor Vessel Water Level - Low Low Low, Level 1
- b. B.1.b. Drywell Pressure - High
- c. B.1.c. LPCI B and C Injection Valve Injection Line Pressure Low Interlock

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- d. B.1.d. LPCI Pump B Start Time Delay Relay
 - e. B.1.e. LPCI Pump Discharge Flow - Low
 - g. B.1.g. LPCI B and C Injection Valve Reactor Pressure Low Interlock
2. Automatic Depressurization System Trip System "B"
- a. B.2.a. Reactor Vessel Water Level - Low Low Low, Level 1
 - b. B.2.b. Drywell Pressure - High
 - c. B.2.c. Initiation Timer
 - d. B.2.d. Reactor Vessel Water Level - Low, Level 3
 - e. B.2.e. LPCI Pump B and C Discharge Pressure - High

NOTE: The Unit 1 Technical Specifications for this item erroneously states "LPCS" instead of "LPCI". The correction of this typographical error is addressed in Appendix IX.

- h. B.2.h. Drywell Pressure Bypass Timer

Division 3 Trip System - HPCS System

- a. C.1.a. Reactor Vessel Water Level - Low Low, Level 2
- b. C.1.b. Drywell Pressure - High
- c. C.1.c. Reactor Vessel Water Level - High, Level 8
- d. C.1.f. Pump Discharge Pressure - High
- e. C.1.g. HPCS System Flow Rate - Low

D. Footnotes

1. On Table 3.3.3-1, change footnote (a) that applies to the MINIMUM OPERABLE INSTRUMENTS column for TRIP FUNCTION D, "LOSS OF POWER", to read footnote (d).
2. On Table 3.3.3-1, add the following as footnote (d) to the TABLE NOTATION section:

"A channel/instrument may be placed in an inoperable status for up to 2 hours during periods of required surveillance without placing the trip system/channel/instrument in the tripped condition provided at least one other OPERABLE channel/instrument in the same trip system is monitoring that parameter."

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These footnote changes are proposed to prevent the AOT associated with Footnote (a) from being applied to the Emergency Bus undervoltage and degraded voltage relays. The LTRs issued by GE and SERs approved by the NRC did not address and do not justify extending AOTs for the undervoltage and degraded voltage relays. Adding the the new footnote serves to prevent misinterpretation.

- E. Technical Specification Bases - Add the following statement to the bases to provide reference for the amendment:

Specified surveillance intervals and surveillance and maintenance outage times have been determined in accordance with NEDC-30936P-A, "Technical Specification Improvement Methodology (With Demonstration for BWR ECCS Actuation Instrumentation)", Parts 1 and 2, December 1988, and RE-025 Revision 1, "Technical Specification Improvement Analysis for the Emergency Core Cooling System Actuation Instrumentation for LaSalle County Station, Units 1 and 2", April 1991. When a channel is placed in an inoperable status solely for performance of required surveillances, entry into LCO and required ACTIONS may be delayed, provided the associated function maintains ECCS initiation capability.

Justification for the Proposed Changes

On November 14, 1985, the BWROG submitted Licensing Topical Report NEDC-30936P, "BWR Owners' Group Technical Specification Improvement Methodology (With Demonstration for BWR ECCS Actuation Instrumentation), Part 1", for NRC review. The Topical provided BWR models and methodology for probabilistic analyses to identify and evaluate improvements to technical specifications for ECCS actuation instrumentation. It included a demonstration analysis using a sample ECCS actuation instrumentation test interval requirement and allowable out-of-service time. The example involved a single technical specification line item of a single instrumentation type. Part 1 of NEDC-30936-P concluded that, from the standpoint of probabilistic considerations, the STI and AOT could be increased with little effect on plant safety. Following engineering review, it was concluded that a change from 31 days to 92 days for STI and a change from 1 hour to 24 hours for AOT would be appropriate.

By letter from Ashok C. Thadani (NRC) to Donald N. Grace (BWROG) dated December 9, 1988, the NRC provided their SER of the NEDC-30936P (Part 1). The NRC concluded in their SER that the

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methodology in the report provides an acceptable generic basis for supporting plant specific technical specification changes related to ECCS. The BWROG subsequently issued the approved version of the Topical, NEDC-30936P-A (Part 1).

On July 23, 1987 the BWROG submitted Licensing Topical Report NEDC-30936P, "BWR Owners' Group Technical Specification Improvement Methodology (With Demonstration for BWR ECCS Actuation Instrumentation), Part 2", for NRC review. This report provides the bases for extending the STI and AOT for all ECCS actuation instrumentation. Similar to the RPS report discussed in Appendix I of this submittal, the analyses documented in NEDC-30936P (Part 2) utilized fault tree modeling to estimate the impact of the proposed changes on the average water injection failure frequency.

The calculation of water injection failure frequency depends on two sets of parameters. The first set consists of initiating events which eventually call for water injection. The second set consists of the probability that the water injection function is unavailable given a demand for injection. During each initiating event, the number of components that are needed for successful completion of the water injection function varies. Therefore, the water injection function unavailability for a given initiating event may differ from that of another initiating event.

A function fault tree was developed for each initiating event in order to quantify the water injection unavailability per demand. The function fault tree modeled the logical relationship of the faults that contribute to water injection unavailability. The function fault tree was used to estimate the water injection unavailability based upon the current Technical Specification requirements and the effect of proposed changes. The results were considered acceptable by the BWROG if the proposed changes resulted in less than a 4% increase in the average water injection failure frequency or if the average water injection failure frequency was calculated to be less than $1E-6$ /year.

The only initiating events studied in this analysis were loss of offsite power (LOSP) initiating events. The LOSP event was chosen for consideration because, based on prior Probabilistic Risk Assessment calculations, LOSP events contribute from 40% to 90% of the calculated core damage frequency for most BWRs. Also, the LOSP analysis is a more severe test of ECCS actuation instrumentation than other accident sequences such as turbine trip, loss of

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EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION

feedwater, and recirculation pump failure. Therefore, the effect of the proposed changes on water injection unavailability and failure frequency for the LOSP initiating event will dominate contributions from all initiating events.

Part 2 of NEDC-30936P concluded that the results of the analysis show that the water injection function unavailability is insensitive to the proposed changes in STI and AOT for ECCS actuation instrumentation. The incremental increase in water injection function failure frequency is either less than 4.0% or less than $1.0E-6$ per year when the STI are extended from 31 to 92 days, AOT for repair are extended from one to 24 hours, and AOT for test are extended from two hours to six hours.

By letter from Charles E. Rossi (NRC) to Donald N. Grace (BWROG) dated December 9, 1988, the NRC provided their SER of the NEDC-30936P (Part 2). The NRC concluded in their SER that the methods and acceptance criteria provided in NEDC-30936P (Part 2) are acceptable for implementation on a plant-specific basis. The BWROG subsequently issued the approved version of the Topical, NEDC-30936P-A (Part 2).

The plant specific evaluation for extending the generic study to the LaSalle ECCS instrumentation configuration was provided by GE in Topical Report RE-025, "Technical Specification Improvement Analysis for the Emergency Core Cooling System Actuation Instrumentation for LaSalle County Station, Units 1 and 2". Any differences between the LaSalle ECCS configuration and the generic one were evaluated to be within the acceptance criteria established in the generic analysis. Revision 1 of RE-025 was issued as an update to the original RE-025 which accounts for modifications which could have affected the original plant specific analysis. It also concluded that the generic basis provided in NEDC-30936P (Part 2) apply to LaSalle.

By letter OG90-319-32D to M. L. Wohl (NRR) from W. P. Sullivan (GE) dated March 22, 1990, "Clarification of Technical Specification Changes Given In ECCS Actuation Instrumentation Analysis", GE clarified Technical Specification of changes given in ECCS actuation instrumentation STI and AOT in Topical Report NEDC-30936P-A. GE stressed that the information presented in the letter was considered in the approved Topical and represent clarifications to ensure that plant specific changes are properly interpreted. The associated changes proposed in this amendment are consistent

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with the information presented in the letter.

Plant modifications installed since RE-025 Revision 1 was issued were evaluated to assess any effects upon the conclusions drawn in the generic and plant specific analyses. The modifications installed do not affect the systems with respect to the extension of STI and AOT. The LaSalle proposal departs from the generic analysis for BWR 5/6 "relay plants" in that the channel functional testing for the manual initiation logic is not being altered from refuel outage to quarterly intervals. This is because functional testing of a divisional ECCS manual initiation relay logic channel requires that the emergency diesel generator and the associated divisional ECCS loop be disabled to prevent initiation. This accordingly requires that the reactor be in a shut down condition.

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CONTROL ROD BLOCK WITHDRAWAL INSTRUMENTATION

The following are the proposed changes to Technical Specifications Section 3/4.3.6, "Control Rod Withdrawal Block Instrumentation":

- A. Surveillance Requirement 4.3.6, add footnote " * " following "CHANNEL FUNCTIONAL TEST" stating that:

"A channel may be placed in an inoperable status for up to 6 hours for required surveillance (or 12 hours for repair) without placing the trip system in the tripped condition provided at least one other OPERABLE channel in the same trip system is monitoring that parameter."

- B. AOT

Table 3.3.6-1 Action 62: Change the AOT from "one hour" to "12 hours".

- C. Channel Functional Test Interval - Technical Specification Table 4.3.6-1. Change the test interval from Monthly to Quarterly for the following trip functions:

1. Rod Block Monitor
 - a. 1.a. Upscale
 - b. 1.b. Inoperative
 - c. 1.c. Downscale
2. APRM
 - a. 2.a. Flow Biased Simulated Thermal Power-Upscale
 - b. 2.b. Inoperative
 - c. 2.c. Downscale
 - d. 2.d. Neutron Flux-High
3. Scram Discharge Volume
 - 5.b. Scram Discharge Volume Switch in Bypass
4. Reactor Coolant System Recirculation Flow
 - a. 6.a. Upscale
 - b. 6.b. Inoperative
 - c. 6.c. Comparator

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CONTROL ROD BLOCK WITHDRAWAL INSTRUMENTATION

- D. Technical Specification Bases - Add the following statement to the bases to provide reference for the amendment:

Specified surveillance intervals and surveillance and maintenance outage times have been determined in accordance with NEDC-30851P-A, Supplement 1, "Technical Specification Improvement Analysis for BWR Control Rod Block Instrumentation", October 1988, and GENE-770-06-1-A, "Bases for Changes to Surveillance Test Intervals and Allowed Out-of-Service Times for Selected Instrumentation Technical Specifications", December 1992. When a channel is placed in an inoperable status solely for performance of required surveillances, entry into LCO and required ACTIONS may be delayed, provided the associated function maintains Control Rod Block capability.

Justification for the Proposed Changes

On June 23, 1986 the BWROG submitted Licensing Topical Report NEDC-30851P, Supplement 1, "Technical Specification Improvement Analysis for BWR Control Rod Block Instrumentation", for NRC review. This Topical provided the basis for extending the control rod block function (CRBF) instrumentation STI from one month to three months. Unlike the analyses discussed in Appendices I and II of this submittal, no specific fault trees were developed for the control rod block instrumentation. Instead, the impact on the average control rod block failure rate was established based upon the results of the analyses presented in Appendix I of this submittal. This approach was taken because the RPS and CRBF share some common instrument inputs. This instrumentation includes Average Power Range Monitors (APRM); Rod Block Monitor (RBM); Reactor Coolant System Recirculation (RR) Flow Sensors; and Scram Discharge Volume (SDV) High Water Level Sensors.

The BWROG report determined that the average control rod block failure rate would increase less than $1\text{E-}4/\text{year}$ (0.06%) from the current failure rate of 0.16/year (based on industry experience). NEDC-30851P, Supplement 1, states that the benefits associated with the proposed changes to the RPS instrumentation offset any potential negative impact of extending the control rod block instrumentation test intervals.

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The NRC issued an SER on September 22, 1988, to extend the STI for CRBF instrumentation following its review and acceptance of NEDC-30851P, Supplement 1. In their SER, the NRC concluded that Supplement 1 provides an acceptable basis for implementing the proposed STI extensions for control rod block instrumentation. The staff also concluded that Supplement 1 did not explicitly address the extension of AOT for control rod block instrumentation.

On February 19, 1991, the BWROG submitted Licensing Topical Report GENE-770-06-1, "Bases for Changes to Surveillance Test Intervals and Allowed Out-of-Service Times for Selected Instrumentation Technical Specifications", for NRC review. This Topical provided the basis for extending the STI and AOT for selected actuation instrumentation systems. The same of type of instrumentation is used to provide the scram discharge volume (SDV) level rod block and the RPS scram signal. From NEDC-30851P-A, the effect on core damage frequency of changing the repair AOT to 12 hours and the surveillance testing AOT to 6 hours was found to be negligible for the RPS. The staff found it reasonable to assume that the increase in rod block unavailability caused by the increases in the AOT for the SDV level input is also negligible, since the rod block instrumentation is similar in type and configuration to that used for the RPS trip. On this basis the NRC accepted the requested changes in AOT for rod block instrumentation.

For the reactor coolant system (RCS) recirculation flow sensors, the increase in AOT for test and repair was not explicitly addressed. However, per NEDC-30851P, increasing the STI from 31 to 92 days does not result in a significant increase in the APRM flow-biased neutron flux scram unavailability and it is further stated that analyses indicated that increasing the test AOT to 6 hours and the repair AOT to 12 hours would produce an even smaller effect on system unavailability. Therefore, if these increases are acceptable for the signal input to the flow-biased APRM trip, similar test and repair AOT increases should also be acceptable for the same type of input to the rod block instrumentation since it should produce a correspondingly insignificant effect on the rod block instrumentation unavailability. On this basis the staff concluded that the requested changes are acceptable.

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GENE-770-06-1 concluded that although the changes to repair and test AOT were not explicitly identified in Supplement 1, the same bases used for changing the STI applies to the AOT changes. This is because the effect of AOT changes is significantly less than the effect of the STI change and is therefore supported by the bases of Supplement 1.

In an SER issued July 21, 1992, the NRC approved GENE-770-06-1. In the SER, the staff concluded that the analyses in it provided an acceptable basis for extending AOT for control rod block instrumentation. In the SER, the staff stated that individual plants must confirm: 1) the applicability of the generic analyses of GENE-770-06-1 to the plant, and 2) that any increase in instrument drift due to the extensions is properly accounted for in the setpoint calculation methodology. LaSalle has reviewed the analyses presented in GENE-770-06-1, and evaluated plant modifications installed since its issuance to assess their applicability on the effects of the conclusions drawn in it. LaSalle concludes that the modifications do not impact the generic analyses of GENE-770-06-1, and that the analyses do indeed apply to LaSalle. The issue of setpoint drift does not apply to LaSalle per the previous discussion.

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APPENDIX V

REACTOR CORE ISOLATION COOLING SYSTEM ACTUATION INSTRUMENTATION

The following are the proposed changes to Technical Specifications Section 3/4.3.5, "Reactor Core Isolation Cooling Actuation Instrumentation":

A. AOT

1. Technical Specification Table 3.3.5-1:

- a. footnote (a): Change the AOT in an untripped condition from 2 (two) hours to 6 (six) hours.
- b. Action 50 a.: Change the AOT in an untripped condition from one (1) hour to 24 (twenty-four) hours.
- c. Action 51: Change "declare the RCIC system inoperable." to "declare the RCIC system inoperable within 24 hours."
- d. Action 52: Change the AOT for an inoperable channel from 8 (eight) hours to 24 (twenty-four) hours.

B. Channel Check - Add requirement for a shiftly channel check of Reactor Vessel Water Level Instrumentation to Functional Unit b. of Table 4.3.5.1-1, Reactor Vessel Water Level - High, Level 8.

C. Channel Functional Test Interval - Technical Specification Table 4.3.5.1-1:

Change the test interval from Monthly to Quarterly for the following trip functions:

- a. - Reactor Vessel Water Level - Low Low, Level 2, and
- b. - Reactor Vessel Water Level - High, Level 8

D. Technical Specification Bases - Add the following statement to the bases to provide reference for the amendment:

Specified surveillance intervals and surveillance and maintenance outage times have been determined in accordance with GENE-770-06-2-A, "Addendum to Bases for Changes to Surveillance Test Intervals and Allowed Out-of-Service Times for Selected Instrumentation

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REACTOR CORE ISOLATION COOLING SYSTEM ACTUATION INSTRUMENTATION

Technical Specifications (BWR RCIC Instrumentation)", December 1992. When a channel is placed in an inoperable status solely for performance of required surveillances, entry into LCO and required ACTIONS may be delayed, provided the associated function maintains RCIC initiation capability.

Justification for the Proposed Changes

In an SER issued December 8, 1988, the NRC staff accepted the analytic methods presented in GE Topical NEDC-30936P Part 1, "BWR Owners' Group Technical Specification Improvement Methodology (With Demonstration for BWR ECCS Actuation Instrumentation)". The effects of extending STI and AOT were determined for one ECCS component at a time, and analyses demonstrated the acceptability of a general methodology for considering technical specification changes for ECCS actuation instrumentation in BWRs. The change in the water injection function unavailability was calculated in the Part 1 report by changing STI and AOT requirements one component at a time. The change was sufficiently small that the average unavailability with respect to time was ignored.

Changes to the STI and AOT for all ECCS actuation instrumentation were proposed in Part 2 of the Topical Report. It was proposed to change all STI from 31 to 92 days, all surveillance test AOT from 2 hours to 6 hours, and all repair AOT from one hour to 24 hours. In an SER issued December 9, 1988, the NRC staff accepted the basis provided in NEDC-30936P Part 2, to extend STI and AOT for ECCS actuation instrumentation.

On February 19, 1991, the BWROG submitted LTR GENE-770-06-2, "Addendum to Bases for Changes to Surveillance Test Intervals and Allowed Out-of-Service Times for Selected Instrumentation Technical Specifications", for NRC review. This Topical provided the basis for extending the STI and AOT for Reactor Core Isolation Cooling (RCIC) actuation instrumentation. The methodology employs data on initiating event frequency, component reliability (including time-dependent and time-independent failure rates, system descriptions, and success criteria). Fault trees were generated for relevant systems, with emphasis on the modeling of the ECCS and RCIC actuation instrumentation. These fault trees were assembled to form the water injection functional fault tree, with RCIC and ECCS unavailabilities as the top event. The water injection

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REACTOR CORE ISOLATION COOLING SYSTEM ACTUATION INSTRUMENTATION

function failure fault tree was evaluated using fault tree quantification codes.

The general acceptance criteria for proposed changes to RCIC actuation instrumentation technical specifications match those proposed and approved for ECCS actuation instrumentation in NEDC-30936P Part 2. They are:

- a. A 4% or less increase in the water injection function unavailability.
- b. In cases where the 4% criterion cannot be met, the impact of the STI and AOT changes is limited to an absolute increase of $1.0E-6/\text{yr}$ in water injection failure frequency.

These criteria were considered acceptable for the RCIC actuation instrumentation review because of the high degree of similarity in form, function, and method of initiation and control exhibited by the two systems.

In an SER issued September 13, 1991, the NRC approved GENE-770-06-2, in which the staff concluded that the analyses in it provided an acceptable basis for extending STI and AOT for RCIC actuation instrumentation. The staff stated that individual plants must confirm: 1) the applicability of the generic analyses of GENE-770-06-2 to the plant, and 2) that any increase in instrument drift due to the extended STI and AOT is properly accounted for in the setpoint calculation methodology. The issue of setpoint drift does not apply to LaSalle per the previous discussion.

Plant modifications installed since the issuance of GENE-770-06-2 were evaluated to assess any effects upon the conclusions drawn in the generic and plant specific analyses. In the SER, the staff stated that individual plants must confirm: 1) the applicability of the generic analyses of GENE-770-06-2 to the plant, and 2) that any increase in instrument drift due to the extensions is properly accounted for in the setpoint calculation methodology. LaSalle has reviewed the analyses presented in GENE-770-06-2, and evaluated plant modifications installed since its issuance to assess their applicability on the effects of the conclusions drawn in it. LaSalle concludes that the modifications do not impact the generic analyses of GENE-770-06-2, and that the analyses do indeed apply to LaSalle. The issue of setpoint drift does not apply to LaSalle per the previous discussion.

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APPENDIX VI

RECIRCULATION PUMP TRIP ACTUATION INSTRUMENTATION

The following are proposed changes to Technical Specifications Section 3/4.3.4, "Recirculation Pump Trip Actuation Instrumentation":

I. ATWS RECIRCULATION PUMP TRIP ACTUATION INSTRUMENTATION

A. AOT

1. LCO 3.3.4.1, Action b.: Change the AOT from 1 (one) hour to 24 (twenty-four) hours.
2. LCO 3.3.4.1, Action c.1.: Change the AOT from 1 (one) hour to 24 (twenty-four) hours.
3. Technical Specification Table 3.3.4.1-1, footnote (a): Change the AOT for a trip system placed in an inoperable status from 2 (two) hours to 6 (six) hours.

B. Channel Functional Test Interval - Technical Specification Table 4.3.4.1-1:

Change the test interval from Monthly to Quarterly for the following trip functions:

- a. Reactor Vessel Water Level - Low Low, Level 2, and
- b. Reactor Vessel Pressure - High

II. END-OF-CYCLE RECIRCULATION PUMP TRIP ACTUATION INSTRUMENTATION

A. AOT

1. LCO 3.3.4.2, Actions b and c.1: Change the AOT in an untripped condition from 1 (one) hour to 12 (twelve) hours.
2. Technical Specification Table 3.3.4.2-1, footnote (a): Change the AOT for a trip system placed in an inoperable status from 2 (two) hours to 6 (six) hours.

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RECIRCULATION PUMP TRIP ACTUATION INSTRUMENTATION

III. Technical Specification Bases

Insert "and surveillance and maintenance outage times" between "Specified surveillance intervals" and "have been determined..." at the bottom of page B 3/4 3-3 to provide reference for the amendment.

Add the following to Bases 3/4.3.4 to provide reference for the amendment:

When a channel is placed in an inoperable status solely for performance of required surveillances, entry into LCO and required ACTIONS may be delayed, provided the associated function maintains the applicable RPT initiation capability.

Justification for the Proposed Changes

On February 19, 1991, the BWROG submitted Licensing Topical Report GENE-770-06-1, "Bases for Changes to Surveillance Test Intervals and Allowed Out-of-Service Times for Selected Instrumentation Technical Specifications", for NRC review. This Topical provided the basis for extending the STI and AOT for EOC-RPT and the ATWS-RPT system trip functions, in addition to other selected actuation instrumentation systems.

The turbine stop valve closure and turbine control valve low hydraulic pressure trips are end of cycle-recirculation pump trip (EOC-RPT) system trip functions initiated by signals common to the RPS. These common trip functions are bounded by NEDC-30851P even though they were not explicitly identified in the generic analysis. GENE-770-06-1 proposed the following for EOC-RPT instrumentation:

Channel functional test frequency change from monthly to quarterly;

AOT for surveillance tests increase from 2 hours to 6 hours; and,

AOT for repair increase from 1 hour to 12 hours.

The ATWS-RPT instrumentation is part of the mitigation system that initiates in the unlikely event of a scram failure. The trip function is initiated by either high reactor pressure or low reactor water level (level 2). The effect of changes to ATWS-RPT

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instrumentation STI and AOT on the reactivity shutdown failure frequency is negligible based on the low RPS failure frequency and the small change in overall ATWS-RPT function unavailability due to STI and AOT changes demonstrated in NEDC-30851P. The negligible change in reactivity shutdown failure frequency is offset by the benefits from reduced inadvertent scrams which is discussed in NEDC-30851P.

GENE-770-06-1 proposed the following for ATWS-RPT instrumentation:

AOT for repair be increased from 1 hour to 24 hours;

Channel functional test frequency be changed from monthly to quarterly;

Trip unit calibration frequency change from 31 to 92 days; and,

AOT for surveillance tests be increased from 2 hours to 6 hours.

In an SER issued July 21, 1992, the NRC approved GENE-770-06-1, and the staff concluded that the analyses in it provided an adequate basis for extending the STI and AOT for EOC-RPT and ATWS-RPT. The BWROG subsequently issued the approved version of the Topical, GENE-770-06-1-A.

In the SER, the staff stated that individual plants must confirm: 1) the applicability of the generic analyses of GENE-770-06-1 to the plant, and 2) that any increase in instrument drift due to the extensions is properly accounted for in the setpoint calculation methodology. In the SER, the staff stated that individual plants must confirm: 1) the applicability of the generic analyses of GENE-770-06-1 to the plant, and 2) that any increase in instrument drift due to the extensions is properly accounted for in the setpoint calculation methodology. LaSalle has reviewed the analyses presented in GENE-770-06-1, and evaluated plant modifications installed since its issuance to assess their applicability on the effects of the conclusions drawn in it. LaSalle concludes that the modifications do not impact the generic analyses of GENE-770-06-1, and that the analyses do indeed apply to LaSalle. LaSalle is not pursuing the trip unit calibration frequency change from 31 days to 92 days. The Reactor Vessel Water Level - Low Low, Level 2 trip function channel calibration is performed on a Refuel interval; The

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RECIRCULATION PUMP TRIP ACTUATION INSTRUMENTATION

Reactor Vessel Pressure - High trip function calibration is performed on a quarterly basis. The issue of setpoint drift does not apply to LaSalle per the previous discussion.

An STI extension for Turbine Stop Valve - Closure and the Turbine Control Valve - Fast Closure trip function testing is not being sought as they have already been changed to quarterly intervals. These changes were proposed in a letter to Dr. T. E. Murley (NRC) from G. G. Benes (ComEd), dated January 28, 1994; Subject: LaSalle County Power Station Units 1 and 2 Request for Exigent Technical Specification Amendment.

The NRC approved the exigent amendment request by letter from A. T. Gody, Jr. (NRC), to D. L. Farrar (CECo), dated February 25, 1994, transmitting amendments No. 95 and 79 to LaSalle County Station Units 1 and 2, respectively.

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APPENDIX VII

MONITORING INSTRUMENTATION

The following are the proposed changes to Technical Specifications Section 3/4.3.7, "Monitoring Instrumentation":

A. AOT

1. On Technical Specification Table 3.3.7.1-1, add footnote "***" following "2/intake" in the MINIMUM CHANNELS OPERABLE column.
2. Add the following to the NOTES section of Table 3.3.7.1-1:

"**A channel may be placed in an inoperable status for up to 6 hours for required surveillance testing without placing the Trip System in the tripped condition, provided at least one other operable channel in the same Trip System is monitoring that Trip Function."

B. Channel Functional Test Interval - Technical Specification Table 4.3.7.1-1:

Change the test interval from Monthly to Quarterly for the Main Control Room Atmospheric Control System Radiation Monitoring Subsystem instrumentation.

C. Technical Specification Bases - Add the following statement to Bases 3/4.3.7.1 to provide reference for the amendment:

Specified surveillance intervals and surveillance and maintenance outage times have been determined in accordance with GENE-770-06-1-A, "Bases for Changes to Surveillance Test Intervals and Allowed Out-Of-Service Times for Selected Instrumentation Technical Specifications", December 1992. When a channel is placed in an inoperable status solely for performance of required surveillances, entry into LCO and required ACTIONS may be delayed, provided the associated function maintains initiation capability.

Justification for the Proposed Changes

On February 19, 1991, the BWROG submitted Licensing Topical Report GENE-770-06-1, "Bases for Changes to Surveillance Test Intervals and Allowed Out-Of-Service Times for Selected Instrumentation Technical Specifications", for NRC review. This Topical provided the basis for extending the STI and AOT for plant systems actuation

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instrumentation trip functions, in addition to other selected actuation instrumentation systems.

Although GENE-770-06-1 did not explicitly discuss BWR/5 monitoring instrumentation, it proposed the following for the BWR/4 Main Control Room Environmental Control System (MCRECS):

- a. Change the channel functional test frequency from monthly to quarterly;
- b. Change AOT for repair from 2 hours to 6 hours; and,
- c. Change AOT for test from 2 hours to 6 hours.

An evaluation in NEDC-31677P included the initiation of the MCRECS as part of the analysis of the primary containment isolation function. The functions of the BWR/4 MCRECS and LaSalle's Main Control Room Atmospheric Control System Radiation Monitoring Subsystem are similar in nature.

In the BWR/4, the control room air inlet radiation monitors measure radiation levels exterior to the inlet ducting of the main control room. A high radiation level may pose a threat to main control room personnel and automatically initiates the MCRECS. The MCRECS is designed to provide a radiologically controlled environment to ensure the habitability of the control room for the safety of plant operators under all plant conditions. The instrumentation and controls of the system warns operators of any abnormal operating transients and automatically initiates action to isolate or pressurize the main control room to minimize the consequences of radioactive material in the environment.

A main control room air intake high radiation high radiation signal is one of several that automatically initiates the MCRECS. Main control room air is recirculated through the charcoal filter with sufficient outside air being drawn in through the normal intake to maintain the main control room slightly positive with respect to the turbine building.

LaSalle's control room HVAC system is common to both Units 1 and 2. It serves the control room (Units 1 and 2), main security control center, and control room habitability storage room. It is designed to maintain a habitable environment and to ensure the operability

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MONITORING INSTRUMENTATION

of all the components in the control room under all station operating conditions.

Outside atmospheric air is used as the source air supply for all subsystems in a normal situation. The radiation monitoring system automatically shuts off normal outside air supply to the system in the event of detecting a high radiation condition from the outside air intake of the control room HVAC system.

Four monitors are provided to detect high radiation at each of two outside air intakes to the control room HVAC system. These monitors indicate radiation levels, and alarm in the control room upon detection of high radiation levels. Emergency makeup air filter trains automatically start upon high radiation signals from two-out-of-four radiation monitors. The emergency makeup air filter trains are designed to limit occupational dose levels below those required by Criterion 19 of 10 CFR 50 Appendix A.

The operability of LaSalle's radiation monitoring instrumentation ensures that the radiation levels are continually measured in the areas served by the individual channels, and that the alarm or automatic action is initiated when the radiation level trip setpoint is exceeded.

LaSalle's Main Control Room Atmospheric Control System Radiation Monitoring Subsystem and the BWR/4 MCRECS provide similar functions, and have similar bases. LaSalle's proposal to change the channel functional test interval from monthly to quarterly, and to invoke the 6 hour AOT for testing is consistent with those proposed (and later accepted) for BWR/4 MCRECS actuation instrumentation.

In an SER issued July 21, 1992, the NRC approved GENE-770-06-1, and the staff concluded that the analyses in it provided an adequate basis for extending the STI and AOT for plant systems actuation instrumentation. The BWROG subsequently issued the approved version of the Topical, GENE-770-06-1-A.

In the SER, the staff stated that individual plants must confirm: 1) the applicability of the generic analyses of GENE-770-06-1 to the plant, and 2) that any increase in instrument drift due to the extensions is properly accounted for in the setpoint calculation methodology. LaSalle has reviewed the analyses presented in

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GENE-770-06-1, and evaluated plant modifications installed since its issuance to assess their applicability on the effects of the conclusions drawn in it. LaSalle concludes that the modifications do not impact the generic analyses of GENE-770-06-1, and that the analyses do indeed apply to LaSalle. The issue of setpoint drift does not apply to LaSalle per the previous discussion.

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APPENDIX VIII

FEEDWATER/MAIN TURBINE TRIP SYSTEM ACTUATION INSTRUMENTATION

The following are the proposed changes to Technical Specifications Section 3/4.3.8, "Feedwater/Main Turbine Trip System Actuation Instrumentation":

- A. Limiting Condition for Operation 3.3.8, change the associated ACTION statements as follows:
 - a. With a feedwater/main turbine trip system actuation instrumentation channel trip setpoint less conservative than the value shown in the Allowable Values column of Table 3.3.8-2, declare the channel inoperable until the channel is restored to OPERABLE status with its trip setpoint adjusted consistent with the Trip Setpoint value.
 - b. With the number of OPERABLE channels one less than that required by the Minimum OPERABLE Channels per Trip System requirement:
 - 1. Within 7 days, either place the inoperable channel in the tripped* condition or restore the inoperable channel to OPERABLE status.
 - 2. Otherwise, be in at least STARTUP within 6 hours.
 - c. With the number of OPERABLE channels two less than required by the Minimum OPERABLE Channels per Trip System requirement:
 - 1. Within two hours place or verify at least one inoperable channel in the tripped* condition, and restore either inoperable channel to OPERABLE status within 72 hours, or,
 - 2. Be in at least STARTUP within the next 6 hours.

Footnote to LCO 3.3.8:

"* An inoperable channel need not be placed in the tripped condition where this would cause the Trip Function to occur."

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B. AOT/STI

1. On Technical Specification Table 3.3.8-1, add footnote "*" following "3" in the MINIMUM CHANNELS OPERABLE PER TRIP SYSTEM column.
2. Add the following to Table 3.3.8-1:

* A channel may be placed in an inoperable status for up to 6 hours for required surveillance testing without placing the Trip System in the tripped condition.

C. Channel Functional Test Interval - Technical Specification Table 4.3.3.1-1:

Change the test interval from Monthly to Quarterly for the Reactor Vessel Water Level-High, Level 8 trip function.

D. Technical Specification Bases - Add the following statement to Bases 3/4.3.8 to provide reference for the amendment:

Specified surveillance intervals and surveillance and maintenance outage times have been determined in accordance with GENE-770-06-1-A, "Bases for Changes to Surveillance Test Intervals and Allowed Out-Of-Service Times for Selected Instrumentation Technical Specifications", December 1992. When a channel is placed in an inoperable status solely for performance of required surveillances, entry into LCO and required ACTIONS may be delayed, provided the associated function maintains Feedwater System/Main Turbine Trip System actuation capability.

Justification for the Proposed Changes

1. Limiting Condition for Operation

The feedwater and main turbine trip instrumentation is designed to detect a potential failure of the Feedwater Level Control System that causes excessive feedwater flow. With excessive feedwater flow, the water level in the reactor vessel rises toward the high water level, Level 8 reference point (L8), tripping the feedwater pumps and the main turbine. This event is considered to be an incident of moderate frequency, which may occur during a calendar year to once per 20 years for a particular plant.

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Three channels of Reactor Vessel Water Level - High, Level 8 instrumentation are provided as input to a two-out-of-three initiation logic that trips the feedwater pumps and the main turbine. A trip of the feedwater pumps limits further increase in reactor vessel water level by limiting further addition of feedwater to the reactor vessel. A trip of the main turbine and closure of the stop valves protects the turbine from damage due to water entering the turbine.

The feedwater and main turbine trip instrumentation is assumed to be capable of providing a turbine trip in the transient analysis for a feedwater controller failure, maximum demand event. The Level 8 trip indirectly initiates a reactor scram from the main turbine trip (above 30% rated thermal power) and trips the feedwater pumps, thereby terminating the event. This limits the neutron flux peak and fuel thermal transient so that no fuel damage occurs. The Minimum Critical Power Ratio (MCPR) remains above the operating limit.

This event is postulated on the basis of a single failure of a control device, specifically one which can directly cause an increase in coolant inventory by increasing the feedwater flow. The most severe applicable event is a feedwater controller failure during maximum flow demand. The feedwater controller is forced to its upper limit at the beginning of the event.

Multiple level sensors are used to sense and detect when the water level reaches the L8 setpoint. At this point in the logic, a single failure will not initiate or prevent a turbine trip signal. Turbine trip signal transmission, however, is not built to single-failure criterion. The result of a failure at this point would have the effect of delaying the pressurization "signature". However, high moisture levels entering the turbine will be detected by high levels in the moisture separators which are designed to trip the unit. In addition, excessive moisture entering the turbine will cause vibration to the point where it too will trip the unit.

Scram trip signals from the turbine are designed such that a single failure will neither initiate nor impede a reactor scram trip initiation. The LCO requires three channels of the Reactor Vessel Water Level - High, Level 8 instrumentation to be OPERABLE to

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ensure that no single instrument failure will prevent the feedwater pump turbines and main turbine trip on a valid Level 8 signal. Two of the three channels are needed to provide trip signals in order for the feedwater and main turbine trips to occur. Each channel must have its setpoint set within the specified Allowable Value of Table 3.3.8-2. The Allowable Value is set to prevent overfilling the reactor vessel which may result in high pressure liquid discharge through the safety/relief valve discharge lines.

The consequences of this event do not result in any temperature or pressure transient in excess of the criteria for which the fuel clad, pressure vessel, or containment are designed; These barriers therefore maintain their integrity and function as designed.

Although this event does not result in fuel failure, normal coolant activity is discharged to the suppression pool via SRV operation. There will be no exposure to operating personnel because this activity is contained in the primary containment. Since this event does not result in an uncontrolled release to the environment, plant operators can choose to leave the activity bottled up in the containment or discharge it to the environment under controlled meteorological and release conditions. If purging of the containment is chosen, the release will have to be in accordance with prescribed requirements. The radiological consequences of this event, at its worst, would only result in a small increase in the yearly integrated exposure level.

With one channel inoperable, the remaining two OPERABLE channels can provide the required trip signal. However, overall instrumentation reliability is reduced because a single failure in one of the remaining channels concurrent with feedwater controller failure, maximum demand event, may result in the instrumentation not being able to perform its intended function. Therefore, continued operation is only allowed for a limited time with one channel inoperable. If an inoperable channel cannot be restored to OPERABLE status within the required time of 7 days, the required action of being in at least STARTUP within the next 6 hours must be taken.

The 7 day timeclock is based on the low probability of the moderate frequency event occurring coincident with a single failure in a remaining OPERABLE channel. Placing the inoperable channel in trip would conservatively compensate for the inoperability, restore capability to accommodate a single failure, and allow operation to

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continue with no further restrictions.

Alternately, if it is not desired to place an inoperable channel in trip, as in the case where placing the inoperable channel in trip (with another channel already in trip) would result in a feedwater or main turbine trip, the required action of being in at least STARTUP within the next 6 hours must be taken.

The trip capability is considered maintained when sufficient channels are OPERABLE or in trip such that the feedwater and main turbine trip logic will generate a trip signal on a valid signal (2 out of 3 logic). This requires two channels to each be OPERABLE, or one being OPERABLE and one being in trip. With the number of OPERABLE channels two less than that required, within two hours at least one inoperable channel must be placed or verified to be in the tripped condition, and either inoperable channel must be restored to OPERABLE status within the 72 hour time frame, otherwise the unit must be in at least STARTUP within the next 6 hours.

These proposed changes to LCO 3.3.8 serve to eliminate a source of confusion to the Operating Staff, and make the structure of the LCO consistent with other instrumentation LCOs previously presented in this proposal.

2. STI/AOT

On February 19, 1991, the BWROG submitted Licensing Topical Report GENE-770-06-1, "Bases for Changes to Surveillance Test Intervals and Allowed Out-of-Service Times for Selected Instrumentation Technical Specifications", for NRC review. This Topical provided the basis for extending the STI and AOT for plant systems actuation instrumentation trip functions, in addition to other selected actuation instrumentation systems.

GENE-770-06-1 proposed the following for the Feedwater/Main Turbine Trip System:

- a. Change the channel functional test frequency from monthly to quarterly; and,
- b. Increase the AOT for test to 6 hours.

The technical specifications for feedwater/main turbine trip system

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applies to those plants that do not have a direct RPS scram from the reactor water level 8 trip.

The effect of plants not having a direct level 8 scram signal on the RPS failure frequency was evaluated in Topical Report NEDC-30851P-A. The slight increase in risk of a potential moisture carry over in the steam lines due to extending feedwater and main turbine level 8 trip instrumentation STI and AOT for test was judged to be offset by the benefits associated with similarly approved STI and AOT for the RPS.

LaSalle's proposal to change the channel functional test interval from monthly to quarterly, and to invoke the 6 hour AOT for testing for Technical Specifications 3/4.3.8 is consistent with those proposed (and later accepted) for BWR/4 Feedwater/Main Turbine Trip System actuation instrumentation.

In an SER issued July 21, 1992, the NRC approved GENE-770-06-1, and the staff concluded that the analyses in it provided an adequate basis for extending the STI and AOT for plant systems actuation instrumentation. The BWROG subsequently issued the approved version of the Topical, GENE-770-06-1-A.

In the SER, the staff stated that individual plants must confirm: 1) the applicability of the generic analyses of GENE-770-06-1 to the plant, and 2) that any increase in instrument drift due to the extensions is properly accounted for in the setpoint calculation methodology. LaSalle has reviewed the analyses presented in GENE-770-06-1, and evaluated plant modifications installed since its issuance to assess their applicability on the effects of the conclusions drawn in it. LaSalle concludes that the modifications do not impact the generic analyses of GENE-770-06-1, and that the analyses do indeed apply to LaSalle. The issue of setpoint drift does not apply to LaSalle per the previous discussion.

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1. Unit 1 and 2 Technical Specifications, page B 2-9, delete "RSCS and" from Bases 2.2.1, Item 2, Average Power Range Monitor.

The Rod Sequence Control System (RSCS) is no longer in use at LaSalle. Use of the RSCS was eliminated following the approval of Technical Specification Amendments No. 88 (Unit 1) and 73 (Unit 2), transmitted by letter of December 4, 1992, from R. J. Stransky (NRR), to T. J. Kovach (ComEd).

The above deletion was not included in the amendment request transmitted by letter of April 2, 1991, from P. L. Piet (ComEd) to US NRC, Subject: LaSalle County Station Units 1 and 2 Application for Amendment to Facility Operating Licenses NPF-11 and NPF-18, Appendix A, Technical Specifications: Deletion of Rod Sequence Control System and Lowering of Rod Worth Minimizer Setpoint; NRC Docket Nos. 50-373 and 50-374.

2. Unit 1 and 2 Technical Specifications, page 3/4 3-7, Table 4.3.1.1-1, "Reactor Protection System Instrumentation Surveillance Requirements", for FUNCTIONAL UNIT 2.a., delete Operational Condition 1 as a condition for which the surveillance of the Average Power Range Monitor (APRM) Neutron Flux - High, Setdown functional unit is required.

Table 3.3.1-1, "Reactor Protection System Instrumentation", on page 3/4 3-2 of Unit 1 and 2 Technical Specifications shows operational conditions 2, 3, and 5 as APPLICABLE OPERATIONAL CONDITIONS for the Setdown functional unit.

Table 4.3.1.1-1 states operational conditions 1, 2, 3, and 5 in the "OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE IS REQUIRED" column for the Setdown functional unit.

This is proposed in order to make Table 4.3.1.1-1 consistent with Table 3.3.1-1 and Bases 2.2.1, Item 2, Average Power Range Monitor. Per the Bases, the APRM Neutron Flux - High, Setdown scram setting provides adequate thermal margin between the setpoint and the Safety Limits for operation at low pressure and low flow during plant startup (power ascension). The margin accommodates the anticipated maneuvers associated with power plant start, where uniform control rod withdrawal is the most probable cause of significant power increase from sources of reactivity input. This function is of concern during a power ascension as opposed to a power reduction, where in an assumed uniform rod withdrawal approach to the trip

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level, the rate of power rise is no more than 5% per minute. The APRM system would be more than adequate to assure shutdown before the power could exceed the Safety Limit. This function remains in effect until the mode switch is placed in the Run (Operational Condition 1) position.

It is therefore appropriate to delete the requirement for performing the surveillance in Operational Condition 1 since its function is bypassed during this mode of operation.

3. Unit 1 and 2 Technical Specifications Page 3/4 3-8, Table 4.3.1.1-1, "Reactor Protection System Instrumentation Surveillance Requirements", footnote (b), change "The IRM, and SRM channels shall be determined to overlap..." to read "The IRM and SRM channels shall be determined to overlap...". This removes the comma (",") which incorrectly follows the term IRM.
4. Unit 1 and 2 Technical Specifications Page 3/4 3-14, Table 3.3.2-1, Action 20 - Change "with the next 24 hours." to "within the next 24 hours."
5. Unit 2 Technical Specifications Page 3/4 3-21, Table 4.3.2.1-1:
 - a. Trip Function A.4.f., Change "RCIC Steam Line Tunnel W Temperature - High" to "RCIC Steam Line Tunnel ▲ Temperature - High".
 - b. Trip Function A.5.a., Change "RHR Equipment Area W Temperature - High" to "RHR Equipment Area ▲ Temperature - High".
6. Unit 1 Technical Specifications, page 3/4 3-23, delete footnote "*" following paragraph 4.3.3.2 and at the bottom of the page. This footnote no longer applies since it is related to a waiver of surveillances required to allow operation of Unit 1 in Cycle 1 until the first refueling outage.
7. Unit 1 Technical Specifications, page 3/4 3-25, Trip Function B.g., add the word "Injection" between "C" and "Valve" so that the function description reads "LPCI B and C Injection Valve Reactor Pressure-Low (Permissive)".

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8. Unit 1 and 2 Technical Specifications, page 3/4 3-27, Table 3.3.3-1, ACTION 31, insert "requirement" following the term "Trip Function".
9. Unit 1 Technical Specifications Page 3/4 3-33, Table 4.3.3.1-1, Item B.2.e: Change "LPCS" to "LPCI".
10. Unit 1 Technical Specifications, page 3/4 3-86, delete the footnote that reads "*The specified 18 month interval may be waived for Cycle 1 provided the surveillance is performed during Refuel 1.". This footnote no longer applies since it is related to a waiver of a surveillance required to allow operation of Unit 1 in Cycle 1 until the first refueling outage.
11. Applicability of the Provisions of Specification 4.0.4

The following footnote is being added to the Reactor Protection System Instrumentation Surveillance Requirements items 1.a., 1.b., and 2.a. of Table 4.3.1.1-1, and Control Rod Withdrawal Block Instrumentation Surveillance Requirements items 2.d., 3.a., 3.b., 3.c., 3.d., 4.a., 4.b., 4.c., and 4.d. of Table 4.3.6-1:

The provisions of Specification 4.0.4 are not applicable for a period of 24 hours after entering OPERATIONAL CONDITION 2 or 3 when shutting down from OPERATIONAL CONDITION 1.

Technical Specification 4.0.4 (verbatim): "Entry into an OPERATIONAL CONDITION or other specified applicable CONDITION shall not be made unless the Surveillance Requirements associated with the Limiting Condition for Operation have been performed within the applicable surveillance interval or as otherwise specified. This provision shall not prevent passage through or to OPERATIONAL CONDITIONS as required to comply with ACTION statements."

The channel functional tests for Reactor Protection System Instrumentation items 1.a., 1.b., and 2.a. of Table 4.3.1.1-1 and Control Rod Withdrawal Block Instrumentation items 2.d., 3.a., 3.b., 3.c., 3.d., 4.a., 4.b., 4.c., and 4.d. of Table 4.3.6-1 are mode switch dependent and cannot be performed in Mode 1. Specification 4.0.4 requires that surveillances as specified in the above Tables be performed prior to entry into an Operational Condition for which the surveillance is required.

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Surveillances that are required for an Operational Condition during plant shutdown shall be conducted prior to entering the applicable Operational Mode or other specified condition. During plant startup the surveillances required for a particular Operational Condition as specified by Technical Specifications must be performed prior to entering that Operational Condition. In the cases cited above in which the complete surveillances cannot be achieved, such as during a plant shutdown, then the required surveillances should be performed to the extent possible prior to changing modes and then completed within 24 hours of entering the Mode or condition in which the surveillance is required. The stabilization of the plant should be given primary consideration.