NUREG-0519
Supplement No. 3

Safety Evaluation Report

related to the operation of LaSalle County Station, Units 1 and 2

Docket Nos. 50-373 and 50-374

Commonwealth Edison Company

U.S. Nuclear Regulatory Commission

Office of Nuclear Reactor Regulation

April 1982

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1 INTRODUCTION AND GENERAL DISCUSSION

1.1 Introduction

On March 5, 1981, the Nuclear Regulatory Commission (NRC) staff issued its Safety Evaluation Report (NUREG-0519) regarding the applicantion by Commonwealth Edison Company (hereinafter referred to as the applicant) for licenses to operate the La Salle County Station, Unit Nos. 1 and 2 (hereinafter referred to as La Salle or facility), Docket Nos. 50-373 and 50-374. On June 12, 1981, the Safety Evaluation Report was supplemented by Supplement No. 1 which documented the resolution of several outstanding issues in further support of the licensing activities. On February 12, 1982, the NRC staff issued Supplement No. 2 to the Safety Evaluation in which we addressed the open items identified in the Safety Evaluation Report and Supplement No. 1. This report is Supplement No. 3 to our Safety Evaluation Report.

In this supplement to the Safety Evaluation Report, we address several items that have come to light since the previous supplement was issued. Also discussed in this supplement is the basis for issuance of a low power operating license (up to 5 percent of full power), see Section 15.4 of this report.

The items addressed in this report are covered in sections having the same number and title as the section of the Safety Evaluation Report and its supplements in which they were previously discussed. Appendix A of this report is a continuation of the chronology of the radiological review of La Salle. Appendix B is a list of the principal NRC staff reviewers who contributed to this supplement. Appendix C lists errata to the Safety Evaluation Report. The NRC project manager for La Salle is Dr. Anthony Bournia. Dr. Bournia may be contacted by writing to the Division of Licensing, U.S. Nuclear Regulatory Commission, Washington, D.C., 20555.

1.9 Outstanding Issues

At this time, there are no outstanding issues that must be resolved prior to issuance of an operating license for La Salle County Station, Unit No. 1 authorizing fuel loading and operation at power levels up to five percent of full power. A number of actions are identified in Attachment 1 to the license that must be completed to the Commission's satisfaction prior to proceeding to certain specified operational modes.

1.10 License Conditions

Since the issuance of Supplement No. 2 to our Safety Evaluation Report, additional issues were identified by the applicant whereby the operating license will be conditioned or a previously identified license condition is modified as a result of our review. The modified and additional items listed below are discussed further in the sections of this supplement as indicated.

(1) Concrete Containment (3.8.1)
(2) Fire Doors and Dampers (9.5.2.2)
(3) Plant Areas Containing Redundant Divisions (9.5.4)
(4) Industrial Security (13.6)

3 DESIGN CRITERIA FOR STRUCTURES, SYSTEMS, AND COMPONENTS

3.8 Design of Seismic Category I Structures

3.8.1 Concrete Containment

The applicant submitted a draft tendon surveillance technical specification for our review. Upon review of the above material, we concluded that the proposed tendon surveillance program is not in accordance with our position. The apparent discrepancy lies in that the applicant determined the predicted losses of the initial prestress on the basis of the average of five vertical and nine horizontal tendons selected from each tendon group for the surveillance. We recommend that when a tendon is selected randomly during an inspection from a group of tendons, its lift-off value should be checked to see if it is within the predicted tolerance band for that tendon. Furthermore, we stipulate that over the 40-year span, the predicted prestressing force, considering high time-dependent losses in any tendon selected for surveillance and allowing for the expected breakage, should not fall below the required design prestressing force at an anchorage. The information provided by the applicant indicates that this requirement is satisfied when the calculations are performed on the basis of the averages of the 14 tendons selected for surveillance but not when the forces and losses are considered for the individual tendons. This does not conform to our position as described above and, therefore, is not acceptable.

By technical specification, we will impose the following and the applicant has agreed:

(1) The lift-off force of tendons shall be monitored on the basis of each individual tendon surveyed instead of the average of the population of the group of tendons surveyed.

(2) When the lift-off force is below the lower bound of the predicted initial prestress force minus the losses which occur between the preoperational structural integrity test and the time of the surveillance test, the applicant will provide an engineering evaluation to demonstrate structural integrity of containment within the time limits stipulated in the Technical Specifications.

In addition, the applicant will provide the predicted lift-off forces according to the tables in the Technical Specifications. These values are not presently contained in the Technical Specifications. We will condition the operating license of Unit 1 that the applicant supplies this information prior to full power.

8 ELECTRIC POWER

8.3 Onsite Power Systems

8.3.1 General Description

8.3.1.1 Alternating Current Power System

In our Safety Evaluation Report, we stated that the diesel generators installed at La Salle were acceptable but noted that the applicant had committed to perform some additional tests at the plant site, as part of the preoperational testing program.

By letter dated March 25, 1982, the applicant reported on the preoperational test results for La Salle Unit 1 "0" and "1A" diesel generators. The results indicated noncompliance with the underfrequency and undervoltage guidelines of Position C.4 of Regulatory Guide 1.9 (Revision 2), "Selection, Design and Qualification of Diesel-Generator Units Used as Standby (Onsite) Electric Power Systems at Nuclear Power Plants." For example, the results indicate an undervoltage of 63.7 percent (2650 volts), an underfrequency of 94.3 percent (56.5 Hz), and a frequency recovery of the load sequence interval of 82.5 percent versus the Position C.4 of the Regulatory Guide 1.9 recommending 75 percent, 95 percent and 60 percent, respectively. The applicant in justifying its diesel-generator design stated the following:

(1) The preoperational test using actual emergency core cooling system loads demonstrated that the diesel generators have the capability to start and accelerate for all loads to rated speed within the required time period without failing.

(2) A number of margin tests demonstrated that the diesel generators can accept a step load increase that is greater than the actual step load and can endure voltage drops of 60, 58.7 and 51.5 percent without experiencing instability resulting in generator voltage collapse or inability of the voltage to recover.

(3) During load sequencing, no loads tripped due to undervoltage.

(4) The voltage recovered to within 10 percent of the nominal voltage in less than 50 percent of the load sequence interval (approximately 2 seconds).

(5) The frequency recovered to within 2 percent of nominal before the next load was applied automatically.

In addition, motors connected to the Class 1E busses, that will be subject to the undervoltage transients, are designed to start and accelerate their loads with terminal voltages at 80 percent. The time that the voltage will be below 80 percent due to the transient is less than one second and will not appreciably reduce the overall reliability of the motors.

Based on the above justification, we conclude that the La Salle diesel-generato design demonstrates the practical compliance with Position C.4 of Regulatory Guide 1.9 (Revision 2) and is acceptable with the following exception. The trip setpoint for the first level of undervoltage protection relays on the Class 1E busses was approached during these preoperational voltage transients. It is our concern that the diesel generator or essential loads may trip on undervoltage. To resolve this concern, we will require that the undervoltage trip setpoint be changed and periodically verified to demonstrate that an undervoltage trip signal will not be generated for a specified time (minimum of 3 seconds) when subject to a 40-percent voltage drop. This requirement is consistent with the Position B.2 of Branch Technical Position PSB-1, "Adequacy of Station Electric Distribution System Voltages." We will incorporate the above requirements into the La Salle Technical Specifications; and, therefore, consider this item resolved.

9 AUXILIARY SYSTEMS

9.5 Fire Protection Systems

In Supplement No. 2 to our Safety Evaluation Report, we indicated that the fire protection review of the La Salle County Station was complete with no open items. However, the applicant in a letter dated March 25, 1982 provided additional information concerning potential deviations from previous commitments in three specific areas:

(1) Coverage of the fixed water extinguishing system for the cable spreading room - Section D.3(a) of the Branch Technical Position ASB 9.5-1, "Guidelines for Fire Protection for Nuclear Plants,"

(2) Unlabeled fire doors and frames - Section D.1.j of the Branch Technical Position - ASB 9.5-1, and

(3) Fire protection for redundant cables in the diesel generator corridor - Section III.G, Appendix R.

Our assessment of the above deviations is presented below.

9.5.1 Description and Evaluation

9.5.1.2 Sprinkler and Standpipe Systems

In our Safety Evaluation Report, we indicated that the applicant had committed to provide a design in conformance with the National Fire Protection Associated Standard (NFPA) No. 15 "Standard for Water Spray Fixed Systems." By letter dated March 25, 1982, the applicant provided additional information on the pre-action sprinkler system installed to protect the cable spreading room. The installed system deviates from NFPA Standard No. 15 because the pattern of the sprinkler discharge when tested did not cover all the areas that need to be protected.

The system is designed with 65-degree spray heads mounted above each cable tray and spaced approximately 10 feet apart. This design is to provide the NFPA standard recommended discharge density over the entire horizontal area of the cable tray. During a demonstration of the sprinkler system, however, it was noted that the spray did not cover the entire ten-foot tray section between spray heads.

The applicant is providing an alternate shutdown train, and, therefore, losing cables in this room will not affect the shutdown capabilities of the plant. Consequently, the principal purpose of the sprinkler system in this area is to control a postulated fire and to prevent its spread to adjacent safety-related areas, and the present system is adequate for this purpose. Therefore, modifications to assure the systems actual performance to completely meet the design criteria of NFPA Standard No. 15 will not provide a substantial increase in the level of safety of the plant.

On this basis, we conclude that the cable spreading room fire protection meets the guidelines of Branch Technical Position ASB 9.5-1 Section D.3(a), and is, therefore, acceptable.

9.5.2 Other Items Related to the Station Fire Protection Program

9.5.2.2 Fire Doors and Dampers

In our Safety Evaluation Report, we indicated that the applicant had committed to provide doors in accordance Branch Technical Position ASB 9.5-1 which requires all doors and frames be labeled by a nationally recognized testing laboratory. In a letter dated March 25, 1982, the applicant indicated that several doors and frames did not meet its commitment. The applicant stated that three problems were identified:

(1) Certain doors have been installed with nonlabeled frames or certain doors are larger than those tested and are, therefore, unlabeled,

(2) Certain double doors have been installed with electric strikes. The electric strikes are only accepted by the testing laboratory for use in single doors, and

(3) Certain fire doors have been replaced with 2½-inch-thick steel doors which have not been tested, and, therefore, unlabelled.

The applicant has compared the design details of the unlabeled doors and frames with those of labeled doors and frames, and concludes that they are equivalent. Our understanding is that labeled doors and frames are fire tested to ensure that the door will remain in place during fire exposure. For door sizes that can be tested, an actual fire test rather than a comparative analysis must be used to demonstrate the fire rating of the unlabelled doors and frames, and, therefore, we cannot find these doors and frames acceptable.

Electric strikes are acceptable on single doors when the strike is mounted on the fixed door frame. The applicant has installed double doors with an electric strike mounted on the inactive leaf. This double door configuration is different than that tested and found acceptable by the testing laboratory. We, therefore, can not find such an installation acceptable.

The applicant has replaced labeled doors with 2½-inch-thick steel doors in some areas for security purposes. There is no adequate assurance that the modified door and frame assemblies will perform as required during fire exposure. This configuration is different than that tested and found acceptable by the testing laboratory. We, therefore, cannot find these doors acceptable.

At our request, the applicant in a letter dated April 7, 1982, has committed that prior to startup after the first refueling outage to either

(1) perform an engineering review of the manufacturer's certified doors and door frames by a nationally recognized laboratory to certify that the door and door frames provide the required fire resistance, or

(2) test a replicate "as installed" door assembly by a nationally recognized laboratory to determine the door rating, or

(3) replace manufacturer's labeled doors and door frames with UL rated items.

With this commitment, we find the fire doors and frames will meet the guidelines of Branch Technical Position ASB-9.5-1, Section D.1.j, and are, therefore, acceptable. We will condition the operating license of Unit 1 to reflect this commitment.

9.5.4 Plant Areas Containing Redundant Divisions

In our Safety Evaluation Report, we stated that the applicant committed to provide an automatic sprinkler system and an automatic detection system in all areas where redundant safety shutdown systems are not separated by a 3-hour fire-rated barrier. In addition, wherever redundant systems are separated by less than 20 feet of clear, open space, one of the redundant systems will be completely enclosed in a 1-hour fire-rated barrier. By letter dated March 25, 1982, the applicant indicated that the fire protection provided in the diesel generator corridor (Fire Zone 5C11) deviates from our guidelines.

Both division power and control cables pass through this area. All cable trays are horizontal; and, particularly, the power cable trays are located within 3 feet from the ceiling. Three inches of kaowool insulation, equivalent to a 1-hour fire barrier, has been installed on the bottom and both sides of the Division II power cables. The top of the tray has been left open to permit heat loss from the cables. The Division I power cables are protected by a pre-action sprinkler system. The Division I power cables are located approximately 6.5 feet below the Division II power cables. This arrangement does not meet our guidelines.

At our request, the applicant in a letter dated April 7, 1982, committed to install, prior to initial criticality, a 1-hour rated fire barrier on all four sides of the power cable tray which previously was only partially insulated, and area sprinkler system in the diesel generator corridor.

With this commitment, we find the fire protection provided for the diesel generator corridor will meet the guidelines in Section III.G of Appendix R and is, therefore, acceptable. We will condition the operating license of Unit 1 to reflect this commitment.

9.6 Other Auxiliary Systems

9.6.3 Diesel-Generator Auxiliary Systems

Our Safety Evaluation Report did not address the quality group classification and design standards to which the diesel-generator auxiliary (fuel oil, cooling water, air starting, lubrication, and combustion air intake and exhaust) systems were designed. In letters dated March 3 and 18, 1982, the applicant provided additional information on these systems.

The applicant stated that all piping and components for the auxiliary systems that are mounted on the engine and on the engine skid are considered engine mounted and are part of the engine package. The engine skid interface was defined by the applicant as the first connection on the engine skid. Therefore, all quality group classification boundaries discussed in Sections 3.2 and 9.5.4 through 9.5.8 of the Final Safety Analysis Report were based on this definition.

The diesel engine auxiliary systems' piping and components up to the diesel engine skid interface, are designed to seismic Category I, American Society of Mechanical Engineers, Section III, Class 3 (Quality Group C) requirements and meet the recommendations of Regulatory Guide 1.26, "Quality Group Classifications and Standards for Water-, Steam-, and Radioactive Waste Containing Components of Nuclear Power Plants"; and Regulatory Guide 1.29, "Seismic Design Classification." The engine skid mounted piping and components, from the engine block to the engine skid-interface, are considered part of the engine assembly and are seismically qualified and/or seismically analyzed to seismic Category I requirements as part of the diesel engine package. This piping and the associated components, such as valves, fabricated headers, fabricated special fittings, and the like are designed, manufactured, and inspected in accordance with the guidelines and requirements of ANSI Standard B31.1, "Code for Pressure Piping"; ANSI N45.2, "Quality Assurance Program Requirements for Nuclear Facilities"; and 10 CFR 50, Appendix B. The engine skid mounted auxiliary systems piping and associated components are intentionally overdesigned (subjected to low working stresses) for the application, and, therefore, resulting in high operational reliability. The design of the engine-mounted auxiliary system piping and components to the cited design philosophy and standards are considered equivalent to systems designed to American Society of Mechanical Engineers, Section III, Class 3 requirements with regard to systems functional operability and inservice reliability.

Based on our review, we conclude that the engine-mounted and engine skid-mounted piping and components of the emergency diesel-engine auxiliary systems (fuel oil, cooling water, air starting, lubrication and air intake and exhaust) meet the requirements of Criteria 2, 4, 5 and 17 of the General Design Criteria, meet the guidance of the cited Regulatory Guides and Standard Review Plans as specified in the Safety Evaluation Report, they can perform their design safety function and meet the recommendations of NUREG/CR-0660 and industry codes and standards, and are, therefore, acceptable.

13 CONDUCT OF OPERATIONS

13.6 Industrial Security

In our Supplement No. 1 to our Safety Evaluation Report, we concluded that the applicant's security plans were acceptable. We also indicated that an ongoing review of the progress of the implementation of these plans will be performed by the NRC staff to assure conformance with the performance requirements of 10 CFR Part 73. As a result of the NRC staff's review and as a result of revisions filed by the applicant, we are conditionally exempting the applicant from certain implementing provisions of the approved security plan by license condition. In addition, equivalent compensatory measures as specified by the applicant's April 1, 1982 letter have been approved until July 1, 1982 as requested by the applicant.

The applicant's security plan is being withheld from public disclosure in accordance with Section 2.790(d)(I) of 20 CFR Part 2.

15 ACCIDENT ANALYSIS

15.4 Relative Risk of Low Power Operation

Section 15 of the Safety Evaluation Report provides an evaluation of the consequences of postulated events that are categorized as "design basis" accidents. These are accidents whose likelihood is judged to be sufficiently large that mitigative steps must be taken to reduce their consequences. The overall risk of operation of a light-water nuclear plant at full power has been evaluated generically by the staff in the Reactor Safety Study (WASH-1400) and other studies. These overall risk evaluations included consideration of the risk to the public due to all accidents, both design basis accidents and very unlikely, severe accidents that are beyond the design basis accidents of Section 15 of the Safety Evaluation Report.

This section of this report is a comparative evaluation of the risk of low power operation of La Salle County Station, Unit 1, compared to the risk of full power operation.

The applicant requested a license to operate the La Salle County Station, Unit 1 up to 5 percent of rated power during its low power testing program. The applicant has stated that the planned period of time at or near 5 percent would be about 14 days. We have examined the reduction in risk associated with this proposed testing program compared to long-term full power operation. The assessment was similar to that conducted for several pressurized water reactors during the past 2 years. There are three major factors which contribute to a substantial reduction in risk for low power testing as compared to equilibirium full power operation. First, there is additional time available for the operators to correct the loss of important safety systems needed to mitigate relatively high risk events, or to take alternate courses of action. Second, the fission product inventory during this time would be very much less than during full power operation. Third, there is a reduction in required capacity for mitigating systems at low power.

Since the publication of the Reactor Safety Study (WASH-1400), the NRC staff and the industry have continued to study the risk to the public from potential severe accidents at nuclear power plants. Although a risk assessment study has not been performed for a BWR-5 (the La Salle class of plants), studies do exist for a BWR-4 (Limerick Generating Station, Units 1 and 2, Docket Nos. 50-352 and 50-353) and a BWR-6 (Grand Gulf Nuclear Station, Units 1 and 2, Docket Nos. 50-416 and 50-417). The results of these studies support the conclusions that the event scenarios dominating accident risks are generally the same for different classes of boiling water reactors.

It was determined for this assessment that the events which dominate risk for a boiling water reactor could be placed in four categories:

(1) Events (both loss-of-coolant accident (LOCA) and non-LOCA) which include reactor scram but failure to remove heat from the containment.

(2) Non-LOCA events which include reactor scram but failure to inject water into the reactor vessel.

(3) LOCAs with failure of the required emergency core cooling systems (ECCS).

(4) Anticipated transients without scram (ATWS) events.

The events in these four categories were examined to estimate the reduction in the probability of the event because of the additional time available during low power operation for the reactor operators to correct the loss of important safety systems or to take alternate courses of action. Similarly, we have calculated the reduced fission product inventory for operation of an initially unirradiated core at 5 percent for 14 days and have determined the reduction in potential public exposure via reduction in potential release magnitudes. Risk is roughly proportional to the fission product of severe accidents (in which the heat sink is lost) and to the fission product inventory in the core.

It is very important to recognize that this review is based on extrapolating the results of studies for similar classes of boiling water reactors. The NRC staff has utilized engineering judgment in estimating the risk reduction numbers. For these reasons, the risk reduction numbers have larger uncertainties than they otherwise might.

15.4.1 Assessment of Categories

Category 1 Events

Following operation at full power, category 1 events will result in suppression pool heatup and boiling. Suppression pool boiling can overpressurize the containment or result in a reduction in pool level such that net positive suction head (NPSH) to the ECCS pumps is lost. Either containment overpressurization or loss of NPSH defeats the role of the suppression pool as the medium for postaccident heat removal.

Following operation at 5 percent for two weeks, failure to remove heat from the suppression pool results in a very slow increase in pool temperature due to decay heat. The capacity of the suppression pool is very large (~1 million gallons) and is considered to have an allowable temperature rise of about 100 degrees Fahrenheit. For those events resulting in transfer of primary system stored energy to the pool, the initial increase in pool temperature is about 50 degrees Fahrenheit. The decay heat load for the next three days would increase the pool temperature by about another 20 degrees Fahrenheit. A 70-degree Fahrenheit increase in pool temperature poses no threat to containment or the ECCS pump NPSH requirements. Because of the time available, there is a high probability that the operator can take corrective actions to restore pool cooling. For this reason and the low fission product inventory, we believe that the risk due to events in category 1 is reduced by at least a factor of 40,000.

Category 2 Events

Following full power operation, category 2 events would result in reactor coolant boiloff, fuel heatup, and finally fuel melting. Following 5 percent power operation for two weeks, the decay heat rate is so low that, even if passive systems heat losses are neglected, several days would be needed to reduce vessel water level to the top of the active fuel region. At this time, decay heat rate is far below normal passive heat losses to the drywell. Hence, drywell cooler operation could stop boiloff. Because of time available, there is a high probability that the operator can take action to correct ECCS malfunctions or use other systems to restore vessel inventory. For these reasons, we believe that the risk due to category 2 events that result in excessive fuel damage and significant radiological release is reduced by at least a factor of 40,000.

Category 3 Events

The most significant events in this category are the transient induced LOCAs in which a safety relief valve is stuck open. Because of the reduced system pressure and temperature in this class of events, passive system heat losses are substantially less than categories 1 and 2. Therefore, boiloff could continue to eventual core melt at 5 percent if some minimal core cooling is not established. For these events, several hours would elapse before core uncovery would begin and several more hours before uncovery of higher powered center core regions would uncover and core damage would occur. Because of the time available, the operator has a high probability of correcting ECCS malfunctions or cooling with alternate systems. For La Salle, only one control rod drive pump would be more than sufficient to remove decay heat. The reactor core isolation cooling system would be available for a while. Boiling water reactor emergency procedures instruct the operator to use other backup systems as well. For these reasons, we believe that the risk due to events in category 3 resulting in excessive fuel damage and significant radiological release is reduced by factors on the order of 1000 to 100,000.

Category 4 Events

For ATWS events, the low initial power results in a slower rate of heatup of the suppression pool and a large decrease in the amount of sodium pentaborate required to take the reactor to a subcritical condition relative to the full power case. It is estimated that about 2 hours operation at 5 percent power would be required to raise the suppression pool bulk temperature to 200 degrees Fahrenheit assuming operation of both residual heat removal heat exchangers. However, less than about 15 minutes operation of the standby liquid control system would be needed to reach a subcritical, hot standby condition. Because of the additional time available to the operators to act to mitigate ATWS events, and the lower fission product inventory resulting from low power operation, we believe that the risk reduction from category four events is on the order of 1,000-100,000.

15.4.2 Conclusions

The above discussion indicates a significant risk reduction during low power testing for each event category. Combining the factors for each category, we

estimate that the overall reduction in risk to the public should be on the order of 2,000 to 200,000, if La Salle is operated at 5 percent power from initial startup for 14 days compared to equilibrium full power operation. This reduction is similar to that previously estimated for several pressurized water reactors.

22 TMI-2 REQUIREMENTS

22.2 TMI Action Plan Requirements for Applicants for Operating Licenses

II Siting and Design

II.K.3 Final Recommendations of Bulletins and Orders Task Force

Item 18 Modification of Automatic Depressurization System Logic--Feasibility for Increased Diversity for Some Event Sequences

Discussion and Conclusions

In our Supplement No. 1 to our Safety Evaluation Report, we indicated that the applicant has committed to implement the logic modification for the automatic depressurization system by either: (1) eliminating the high drywell pressure trip, or (2) bypassing the drywell pressure trip after runout of a timer started at the low pressure emergency core cooling system initiation. We indicated that either of these modifications were acceptable to us. However, since the issuance of this supplement, the BWR Owners Group, of which the applicant is a member, has indicated that the previously proposed automatic depressurization system logic modifications may unduly complicate operator actions during an anticipated transient without scram event. Consequently, the BWR Owners Group has requested an extension of the schedule for implementation of the above modifications in order to prepare supplemental information on alternative logic designs.

We have reviewed the analysis results and the discussion of the above modifications relative to anticipated transients without scram, and agree with the conclusions of the BWR Owners Group. We require, therefore, the BWR Owners Group new alternative design modifications by October 1, 1982, and prior to startup after the first refueling outage, we will require the applicant to implement the approved new alternative modification. This will be reflected in an operating license condition for Unit 1. In the interim, should rapid vessel depressurization be required due to a break outside containment or a stuck-open relief valve, manual actuation of the automatic depressurization system can be accomplished. Therefore, the delay of implementation is acceptable.

APPENDIX A

CONTINUATION OF CHRONOLOGY FOR THE LaSALLE COUNTY STATION

February 8, 1982	Letter from applicant concerning SQRT Status Report, Volume 12.
February 11, 1982	Letter from applicant concerning Response to Questions on Process Control Program.
February 11, 1982	Letter from applicant concerning Independent Design Review.
February 19, 1982	Letter from applicant concerning Response to Questions on ADS Valves NUREG-0737, Item II.K.3.28.
February 22, 1982	Letter from applicant concerning Proposed Technical Specification changes.
February 22, 1982	Letter to applicant concerning Issuance of Supplement No. 2 to the Safety Evaluation Report.
February 22, 1982	Letter from applicant concerning Emergency Preparedness Excercise Description of Objectives.
February 23, 1982	Letter to applicant concerning Review of LaSalle's Independent Design Verification Program.
February 24, 1982	Letter from applicant concerning Revision of Commitment to 6-Hour Breathing Apparatus.
February 24, 1982	Letter from applicant concerning Fuel Seismic and Loss-of-Coolant Accident Loadings.
February 24, 1982	Letter from applicant concerning Response to Informal Questions on ODCM.
February 25, 1982	Letter from applicant concerning Fire Stops for Non-segregated Phase Bus Duct Penetrations.
March 2, 1982	Letter from applicant requesting for Extension of NRC Materials License SNM-1833.
March 2, 1982	Letter from applicant requesting Unit 2 Construction Permit Extension.
March 8, 1982	Letter from applicant concerning Unit 1 Technical Specification on Tendon Surveillance.
March 8, 1982	Letter from applicant concerning Process Control Program.

March 8, 1982	Letter from applicant concerning Unit 1 Update Revision for Preservice Inspection Report.
March 9, 1982	Letter from applicant concerning NUREG-0737, Item II.K.3.18, ADS Logic Modification and ATWS.
March 9, 1982	Letter from applicant concerning Piping Vibration Monitoring.
March 10, 1982	Letter from applicant concerning Offsite Dose Calculation Manual.
March 10, 1982	Letter from applicant concerning Process Control Program.
March 12, 1982	Letter to applicant requesting for additional information concerning La Salle's Containment Purge/Vent Operation.
March 12, 1982	Letter from applicant concerning Amendment No. 60 to the Final Safety Analysis Report.
March 16, 1982	Letter from applicant concerning Diesel Generator Piping Convention.
March 16, 1982	Letter to applicant concerning Independent Design Review Initial Status Report for the Period of February 11 through March 12, 1982.
March 18, 1982	Letter from applicant concerning Additional Information on Diesel Generator Piping.
March 19, 1982	Letter from applicant concerning Deletion of Topical Report CE-1-A from FSAR Chapter 17.
March 19, 1982	Letter from applicant concerning Deletion of Technical Specifications from FSAR Chapter 16.
March 24, 1982	Letter to applicant concerning Acceptance of Revision 11 to La Salle's Security Plan.
March 24, 1982	Letter from applicant concerning Evaluation of 6 Months of Data from the 10 Meter Meteorological Measurements System.
March 25, 1982	Letter from applicant concerning Diesel Generator Starts.
March 25, 1982	Letter from applicant concerning ESF Divisions 1 and 2 Diesel Generators' Voltage and Frequency Requirements during the ECCS Loading Sequence.
March 25, 1982	Letter from applicant concerning Associated Piping and Engineering Corporation Alteration of Radiographs of Welds in Piping Subassemblies.

March 25, 1982	Letter from applicant concerning Technical Specification on Tendon Surveillance.
March 25, 1982	Letter from applicant concerning Fire Protection.
March 26, 1982	Letter from applicant concerning Comments on Draft License.
March 29, 1982	Letter from applicant transmitting the Process Control Program.
March 29, 1982	Letter from applicant transmitting Technical Specification 3.7.9 Snubbers Revised List for Table 3.7.9.2.
March 30, 1982	Letter to applicant concerning Deletion of Technical Specifications from FSAR Chapter 16.
March 30, 1982	Letter from applicant concerning Security Plan.
March 30, 1982	Letter from applicant concerning Appendix R, Section III.H - Breathing Air Supply.
March 30, 1982	Letter from applicant concerning Item III.A.2 Improving License Emergency Preparedness Long Term, Meteorological and Dose Assessment Capabilities.
March 31, 1982	We meet with representatives from Commonwealth Edison Company in Bethesda, Maryland to discuss the 2.206 Petition submitted by the Attorney General, State of Illinois. (Meeting summary dated April 7, 1982.)
April 1, 1982	Letter to applicant accepting the ODCM and PCP for La Salle County Station Units 1 and 2.
April 1, 1982	Letter from applicant concerning Independent Design Review.
April 1, 1982	Letter from applicant concerning Security Plan.
April 1, 1982	Letter from applicant concerning Deferral of LSCS Preoperational Test Deficiencies Beyond Fuel Load.
April 5, 1982	Letter from applicant indicating Completion of Vendor Review of Emergency Procedures.
April 5, 1982	Letter from applicant concerning Update Unit 2 Preservice Inspection Report.
April 7, 1982	Letter from applicant concerning Fire Protection.
April 8, 1982	Letter from applicant concerning Additional Basis for Forty-Year Operating License.

April 8, 1982	Letter from applicant concerning Revised Reactor Vessel Material Surveillance Program Withdrawal Schedule.
April 9, 1982	Letter from applicant concerning Diesel Generator.
April 13, 1982	Letter from applicant concerning Unit 1 Snubbers.
April 14, 1982	Letter to applicant relative to deferred resolution of preoperational test deficiencies.
April 17, 1982	Letter to Attorney General of Illinois relative to rebar damage and deficiencies of off-gas building roof.

APPENDIX B

NRC STAFF CONTRIBUTORS

This supplement to the Safety Evaluation Report was prepared by the NRC staff. The NRC staff members below were principal contributors to this report.

NRC Staff	Title	Affiliation
S. P. Chan	Senior Structural Engineer	Structural Engineering Branch
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APPENDIX C

ERRATA TO THE SAFETY EVALUATION REPORT

Page 22-13 Lines 36
thru 39
and
Page 22-14 Line 1 - Delete:

(1) An individual should not be permitted to work more than 12 hours straight (not including shift turnover time).

(2) There should be a break of at least 12 hours (which can include shift turnover time) between all work periods.

(3) An individual should not work more than 72 hours in any 7-day period.

(4) An individual should not be required to work more than 14 consecutive days without having 2 consecutive days off.

and replace by:

(1) An individual should not be permitted to work more than 16 hours straight, excluding shift turnover time.

(2) An individual should not be permitted to work more than 16 hours in any 24-hour period, nor more than 24 hours in any 48-hour period, nor more than 72 hours in any seven-day period, all excluding shift turnover time.

(3) A break of at least eight hours should be allowed between work periods, including shift turnover time.

(4) The use of overtime should be considered n an individual basis and not for the entire staff on a shift.

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7. AUTHOR(S)

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MONTH	YEAR
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9. PERFORMING ORGANIZATION NAME AND MAILING ADDRESS (Include Zip Code)
U.S. Nuclear Regulatory Commission
Office of Nuclear Reactor Regulation
Washington, D.C. 20555

DATE REPORT ISSUED

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16. ABSTRACT (200 words or less)

Supplement No. 3 to the Safety Evaluation Report of Commonwealth Edison Company's application for licenses to operate its La Salle County Station, Units 1 and 2, located in Brookfield Township, La Salle County, Illinois has been prepared by the Office of Nuclear Reactor Regulation of the U.S. Nuclear Regulatory Commission. This supplement addresses several items that have come to light since the previous supplement was issued.

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