STATE OF ILLINOIS)
)   SS.
COUNTY OF LASALLE)

AFFIDAVIT

I, [redacted] being fully sworn and under oath do state:

I reside at [redacted] Illinois. I was employed as a core driller at the Commonwealth Edison LaSalle County nuclear plant construction site from approximately June, 1978 until July, 1980.

From about June, 1978 until about February 1980 my employer was Commercial Concrete Sawing and Drilling Company. My duties were the drilling of holes in concrete. I drilled holes ranging in diameter from 1/4" to 3/4" with a small hand drilling machine. I also drilled larger holes, ranging in diameter from 1-1/2" to 8", with a large boring machine. Anchor bolts for the small holes were used for hanging conduit, cable trays, and other electrical equipment. The large holes were used to carry conduit through walls and floors.

I performed core drilling in all buildings, at all elevations, throughout the plant site, including the reactor buildings for Units 1 and 2. During most of the year 1979, my partner and I were assigned primarily to the two reactor buildings. We drilled at all elevations in the reactor buildings. During the time of my employment with Commercial Concrete Sawing and Drilling, I received my

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drilling instructions orally from my foremen, [redacted] a Commercial Concrete employee. My work was observed by [redacted] a superintendent employed by the general contractor, Foley Electrical Co.

From the time I began drilling at LaSalle in June, 1978 until about February, 1980, it was the usual practice, upon contacting metal reinforcement or rebar during core drilling, to drill through the metal rebar. I was instructed to follow this practice, and to the best of my knowledge, it was the general practice among the other core drillers. Occasionally we were instructed to stop and relocate the holes when metal was contacted. But during most of the time period, we cut through the metal.

Small holes were of the following sizes:

Diameter (inches)	Depth (inches)
1/4	1-1/4
3/8	4
1/2	4-1/2
5/8	5
3/4	6

When rebar was contacted in drilling small holes with diameters of 1/2 to 3/4 inches, we changed to the larger "wet drill", which was a boring drill with a carbon bit and a water spray. The wet drill was too large for 1/4 and 3/8 inch holes. When rebar was contacted in drilling these smallest holes, we relocated the holes. My partner and I drilled hundreds of small holes per week. We contacted and

cut through rebar in less than 1/4 of those holes. During the latter part of 1979, each small hole cut through rebar was marked with paint or a felt pen. I am unable to estimate the number of rebar cuts more precisely.

Large holes, ranging in diameter from 1-1/2 inch to 8 inches, were all cut with the large boring machine The depths of the large holes equalled the thickness of the walls, floors, or ceilings through which the holes were cut, and ranged from about 1 foot to 6 feet. For the largest diameters, the bits were carbon with diamond chips. During my employment in 1978-79, core drillers were instructed to cut through rebar when it was contacted during the drilling of large holes We seldom failed to contact rebar with the large boring machine My partner and I followed this practice; and to the best of my knowledge the other drillers did also. Until the end of 1979, I believe that all of the large holes were drilled by Commercial Concrete employees.

When I worked in reactor buildings for Units 1 and 2 during 1979, I drilled large holes through the walls between the two reactor buildings, between the reactor and the off-gas building, and between the reactor and auxilliary building. Large holes were cut at a rate of about foot per hour through concrete. When rebar was contacted and cut took longer.

The locations for holes to be drilled were laid out by employees of Foley Elc. To the best of my knowledge metal

detectors were not used during the first 7-8 months of my employment for locating the holes. A Quality Assurance inspector inspected my work beginning several months after I started working.

I can recall two specific incidents concerning the drilling of large holes through metal rebar. On one occasion I drilled a 6" diameter hole through rebar in the reactor building of Unit 1, at an elevation below 710'. It was at a place where all the steel tied together, and I removed about 25-40 pounds of steel. It took me 2 or 3 days to drill this hole. [redacted] instructed me to keep drilling this hole, and [redacted] added, "If you can't do it, we'll get someone who can."

On a second occasion I drilled a 7" diameter hole in the reactor building of Unit 1 at elevation 735. I hit the 2" rebar, and as I continued to drill the rebar was splitting. I asked [redacted] and [redacted] if I could relocate the hole. [redacted] said, "No." That hole was drilled to a depth of 6 to 7 feet, where we hit a beam in the floor of a room where steam pipes were located. This hole was later grouted in, because it was improperly located.

We filled out a written report, or drill sheet, on each hole we drilled for both small and large holes. The reports showed the location, depth, and diameter of each hole. They also showed whether rebar was contacted or cut.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

APR 15 1982

Docket No. 50-373/374

MEMORANDUM FOR: Harold R. Denton, Director
Office of Nuclear Reactor Regulation

FROM: Richard C. DeYoung, Director
Office of Inspection and Enforcement

SUBJECT: RESULTS OF INDEPENDENT ASSESSMENT OF ALLEGATIONS BY T. FAHNER, ESQ., REGARDING THE LASALLE STATION OFF-GAS FILTER BUILDING

As discussed in our conference call with Messrs. Case, Stello and Keppler on March 29, 1982, IE accepted responsibility for certain actions necessary to respond to the March 24, 1982 filing by T. Fahner, Esq., Attorney General, State of Illinois. Specifically, IE was to conduct an independent assessment of actions taken by RIII on the issues related to the off-gas filter building and to independently assess the allegations related to the off-gas filter building structural adequacy.

We have completed both aspects of this assessment, the details of which are contained in two separate reports. Enclosure 1 (Inquiry and Assessment of RIII Actions) will not be finalized until April 16 but its basic conclusions are reflected in this memorandum. You will be provided a copy upon completion. These reports were prepared on the basis of documentation review, interviews, field measurements and field observations.

Our conclusions as a result of this independent effort are as follows:

1. Region III followed a logical course of action in responding to the allegations related to the off-gas filter building. They first determined whether the allegations pertained to safety related structures, systems or components. This was completed using the reference document of the Safety Analysis Report plus the expertise of RIII staff in BWR plant systems including the off-gas system. Since the allegations were found to relate to non-safety items, no further action was taken. Our independent assessment is that the allegations did not require additional RIII follow-up. This is due to the fact that the off-gas filter building is a non-safety related building which contains systems and equipment which have no function in preventing or mitigating accidents or accident conditions.

CONTACT: R. E. Shewmaker, IE
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Further it was noted during this assessment that when RIII became aware of the allegation of the cutting of reinforcing steel in safety-related structures, prompt action was taken by the investigator to alert his supervision which, in turn, alerted the RIII engineering group. Currently, RIII is working on this matter in conjunction with NRR.

2. The reinforced concrete roof of the off-gas filter building has slight deviations from the thickness of the 12-inch thick slab as specified by the design drawings. The range of deviations of +1-1/4" and -13/16" will have no significant effect on structural behavior.

3. The loading imposed by the temporary construction related transformer on the roof of the off-gas filter building did not exceed the design loads and did not cause structural damage.

4. The current condition of the roof of the off-gas filter building, which includes existing cracks, embedments, anchor bolts and nicked reinforcing steel, is acceptable. There is every reason to expect that the roof system can fully carry the design live load of 100 psf and remain in a serviceable condition while performing its intended function over its service life.

If there are any questions on this assessment and its related details, my staff and I will discuss them with you. We are maintaining the various documents which support this assessment and its conclusions.

RC DeYoung

Richard C. DeYoung, Director
Office of Inspection and Enforcement

Enclosures:
1. Inquiry and Assessment of RIII Actions
2. Assessment of Off-Gas Filter Building

cc w/enclosures: See page 3

cc w/enclosures:
J. H. Sniezek, IE
E. L. Jordan, IE
R. Fortuna, IE
D. G. Eisenhut, NRR
R. A. Purple, NRR
A. Schwencer, NRR
R. H. Vollmer, NRR
J. G. Keppler, RIII
R. L. Tedesco, NRR

ASSESSMENT OF THE OFF-GAS
FILTER BUILDING AT LASALLE
NUCLEAR STATION

by R. E. Shewmaker, P.E.
April 8, 1982

Background:

At noon on March 30, 1982, I was provided with some preliminary information related to statements contained in a petition dated March 24, 1982 from Mr. Fahner, Esq. addressing the off-gas filter building roof by E. L. Jordan. I was alerted that it might be necessary to go to the LaSalle facility to view the structural components in question that same week and provide a written assessment by mid-week the following week.

At the direction of Mr. E. L. Jordan, mid-morning on March 31, 1982, I was instructed to assess in the field (1) whether the reinforced concrete roof of the off-gas filter building met the design requirements (that is, does the as-built condition conform to the drawings) and (2) whether the reinforced concrete roof of the off-gas building can meet its service requirements. The need for such an assessment was apparently the subject of a telephone conference call on March 29, 1982 regarding a petition filed by the Attorney General, State of Illinois requesting a Show Cause Proceeding or Other Relief related to this and other issues. This conference call was followed by a written request for assistance from IE by the RIII Regional Administrator (see Attachment 1).

Initial In-Office Effort (to determine requirements)

During the afternoon of March 30, 1982, I proceeded to review the pertinent portions of the LaSalle Final Safety Analysis Report (FSAR) to determine what the licensee had defined the structural safety requirements to be for the off-gas filter building.

Section 3.2, Classification of Structures, Components and Systems, was examined along with Table 3.2-1 which provides a detailed classification of various plant structures, equipment and components. As noted in the text of Section 3.2, plant structures, systems and components important to safety are designed to withstand the effects of a safe shutdown earthquake and remain functional. These are known as Category I and include all such items if they are required to ensure:

a. The integrity of the reactor coolant pressure boundary,

b. The capability to shut down the reactor and maintain it in a safe condition, or

ENCLOSURE 2

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c. The capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures in excess of the guideline exposures of 10 CFR 100.

This revealed that all of the equipment housed in the off-gas filter building which is part of the off-gas system is classified as seismic Category II as well as the building itself. This means that the equipment and structure are not required to meet the requirements of 10 CFR 50, Appendix B, Quality Assurance Requirements. The quality group classifications for the various portions of the off-gas system are either C or D as indicated in Table 3.2-1 and defined in Regulatory Guide 1.26 and meet the various quality standards of the pertinent industry codes and standards (see Attachment 2).

The FSAR was then reviewed to determine what documented design requirements had been committed to by Commonwealth Edison for the design and construction of the off-gas filter building. The FSAR in Section 3.8.4 describes the criteria for the seismic Category I structures other than containment which in this case did not include the off-gas filter building. Commonwealth Edison provided for another level of safety-related structures in the design criteria known as "Non-Seismic Category I Safety Related Structures," but it also did not include the off-gas filter building. Therefore, no defined criteria existed in the licensee's application which is consistent with the fact that the building was not classified as safety related.

The Safety Evaluation Report (SER) for the LaSalle Nuclear Station was also examined to determine if the NRC staff had accepted the classification of structures, components and systems provided by the applicant. Section 3.2.1 indicates that with the exception of the classification of cooling loop of the spent fuel cooling and cleanup system all structures, components were correctly classified by Commonwealth Edison. In Section 11.2.2 which addresses gaseous wastes the NRC staff specifically stated that the off-gas system is located in the off-gas filter building which is a nonseismic Category I structure. The NRC staff further stated that the process off-gas system and the structure housing the system were acceptable (see Attachment 3).

I also had a discussion with an IE BWR systems engineer concerning the proper classification of "an off-gas filter building and the off-gas system." He indicated that the system was not used for accident prevention or mitigation and that the system would therefore most likely be classified as non-safety related.

On March 31, 1982, I discussed the proper classification of the off-gas system with an IE health physicist who reviewed the typical system's design and function as well as pointing out that many BWR's operate without such a system though newer plants are installing such treatment systems to conform to the ALARA guidelines of 10 CFR 50 Appendix I. His conclusion was that building failure should be of no real concern from a radiological viewpoint.

Also on March 31, 1982, during a meeting held with Commonwealth Edison at the request of the NRC for which a transcript was made, it was reiterated that the

NRC, specifically NRR, had considered the off-gas filter building as a "non-safety grade building" which contained no Category I safety related equipment. In addition, the NRC Region III office also stated that they had treated this as a Category II, or non-safety related building. The licensee stated, however, that they did apply the safety-related quality assurance program to the construction of the off-gas filter building (see Attachment 4).

In conclusion, all information available and assessments made indicated that the licensee's classification of the off-gas filter building was in fact correct in that it is not a Category I structure and that the structural requirements governing the design and construction would be those specified by the owner and his agents and would not necessarily incorporate any NRC requirements. With such a classification, the off-gas filter building would not be part of those structures that would be inspected by the NRC.

Field Effort (to determine as-built conditions):

On April 2, 1982 I visited the LaSalle Nuclear Station facility to obtain information and make first hand observations of the off-gas filter building roof. Three specific concerns were to be addressed during this field effort:

1. Facts related the off-gas filter building reinforced concrete slab roof thickness,

2. Facts related to the external loading of the roof by an electrical transformer, and

3. Facts related to the current condition of the reinforced concrete roof system such as holes, anchor bolts, embedments and cracking.

Roof Thickness

The as-designed roof slab thickness shown on the design drawings is 12" (ref. S&L Dwg. S-188, Rev. J). The top of the concrete was established by design at Elevation 726'-6".

Copies of the surveyors field notes for the surveys which the licensee had completed were obtained. These notes reflect three separate surveys with the first being a single point thickness established on March 10, 1982. This established a total thickness of 1'-2¼" which included 1-3/4" to 2" of insulation, asphalt and gravel or a 12¼" or 12½" concrete roof slab. Two separate surveys were completed on March 29, 1982 by a four-man survey party using Level No. 2R728. W. Larson was the survey party chief. The party was made up of personnel of Walsh Construction Company and are classified as Technical Engineers and are union personnel. Their survey work was performed at the request of Mr. D. Shamblin, Construction Engineer, Commonwealth Edison. His directions were to establish the thickness of the roof slab between the integral roof beams. There are five (5) spans of roof slabs so the team selected three (3) points on approximately each of the spans' centerlines, making a total of fifteen (15) points on which to establish slab thickness. The approximate locations of the

points were established from the column and beam centerlines for reference. The survey party conducted the field work without direct oversight by Commonwealth Edison representatives or quality assurance personnel. No independence from the construction company who built the roof existed.

From a previously established benchmark elevation on the off-gas filter building wall a series of vertical elevations were established by a surveyor's level on the outside of the roof and similarly for the ceiling inside the building at the fifteen (15) points. In the first case the level rod was held at each of the 15 locations on top of the builtup roofing gravel so that the thickness determined at each point reflected the thickness of the concrete roof slab plus the insulation, asphalt and gravel. The concrete roof slab thickness was then determined indirectly by subtracting the approximate design thickness of the insulation, asphalt and gravel which was taken as 2-3/4" to 2". In the second case a 6" steel spike was driven in the outside of the roof to puncture through the gravel, asphalt and insulation until the top of the concrete roof slab was struck. The top of this steel spike was then the point on which the level rod was held at each of the 15 locations on top of the roof. The thickness of the concrete roof slab was then obtained directly from the difference in elevations for the top and the bottom of the concrete roof slab.

From the first case the thickness at one (1) point was determined to be 11.68" assuming a 2" thick built up roof. In the second case the thickness at three (3) points was found to be 11.16", 11.52" and 11.64." I consider measurements of 11.88" as within the allowable measuring errors of ±0.01'. Four (4) measurements of 11.88" were obtained.

The reduction of field survey data has been checked to verify the determination of the thickness of the concrete roof slab based on the field data. No discrepancies were noted. No direct measurements of the concrete roof slab thickness were physically possible since no holes are open through the roof thickness. From the design drawings and field observation the six (6) penetrations through the roof are all sealed. They consist of three (3) roof drains, an HVAC vent/intake, an electrical conduit and an abandoned 12" diameter sleeve which is sealed closed.

Discussions with site personnel revealed that during drilling for concrete anchors there was an instance where drilling into the underside of a 24" thick reinforced concrete floor resulted in penetrating the trap of the floor drain for the 690'-0" floor level in the off-gas filter building. This allowed the water in the trap to flow down through the hole and the possibility of daylight being seen up through the drain via the hole. This occurred in the next floor level below the off-gas filter building's roof.

Based on my review of the facts my assessment is that the reinforced concrete slab portion of the off-gas filter building roof is nominally a 12" thick section with the average thickness, based on fifteen (15) measurements, being 1'-0½". The actual range of values for the fifteen (15) measurements was +1¼" and -13/16" indicating the tolerances are somewhat outside the generally accepted values of +½" and -¼" as provided in ACI 301, specifications for

Structural Concrete for Buildings, 1972, and the values of +3/8" and -½" as provided in the Proposed ACI Standard ACI 117, Tolerances for Concrete Construction and Materials, August 1980. These slight deviations of tolerance will have no significant effect on structural behavior.

External Loading of Electrical Transformer

Based on the information available at the site from Mr. D. Shamblin, Construction Engineer, Commonwealth Edison, there was a transformer placed on the roof of the off-gas filter building to provide electrical service for construction sometime probably in the last half of 1976. The concrete in the off-gas filter building was placed in November of 1975. The size of the base of the transformer was 4'-0" by 13'-2" and the assembly had a total weight of 6700 lbs. The transformer was removed sometime in 1981.

One end of the transformer was placed on the east wall (known as Ab) which is a 12" reinforced concrete wall with the long axis of the tranformer running in the east-west direction nearly aligned over the centerline of the roof beam just north of Column line 13. Based on my calculations this loading, conservatively assuming no loads are directly transmitted to the wall and that the roof slab and beam system must carry the load, results in a value of only about one-third (1/3) of the design live load (100 psf) existing over about 40% of the span of the beam. Therefore, the loading of the construction transformer placed as it was on the reinforced concrete roof of the off-gas building represented less than one-sixth (1/6) of the design live load on the supporting beam.

In addition to this assessment of actual loading versus design capability, I examined the underside of the reinforced concrete off-gas filter building roof in this area for indications of distress that could be caused by excessive loading or underdesign as a result of construction deficiencies. No evidence of load distress were found. Minor hairline cracks were visible but in no greater concentration than elsewhere on the structure.

Based on these facts my assessment is that the temporary construction loading of the electrical transformer was well within the design loads for the reinforced concrete roof of the off-gas filter building and that no structural distress was caused by the loads.

Current Roof Condition

In September of 1979, nearly four (4) years after the concrete had been placed for the roof of the off-gas filter building, the Quality Assurance group of Commonwealth Edison noted some surface cracking in the bottom surface of the off-gas filter building roof slab. The general area was noted as having a high density of expansion anchors and some concern was expressed as to whether the cracking was serious and whether it at all related to the anchors. The area in question was examined by the Walsh Construction Company Quality Assurance Supervisor and the General Superintendent as well as the Commonwealth Edison Company Structural Engineer. The decision was made to chip out two of the

cracks over some several feet to determine the depth of cracking. After chipping the area was patched when the conclusion was reached that the cracks were surface type cracks and no further action would be required.

During the March 31, 1982 meeting mentioned previously, a representative of Commonwealth Edison indicated that the cracks did not exceed one-quarter (¼) inch depth. The cracking was also characterized as shrinkage cracking associated with the slab type construction (see Attachment 4, pp. 15 and 16).

On April 2, 1982, I examined the underside surfaces of the four (4) main roof beams and the five (5) roof slab areas for cracking, holes embedments, anchor bolts and patches. The area where the largest crack size was found consisted of the slab area adjacent to the two (2) patched areas which were repaired in 1979 as a result of the licensee's evaluation of cracking. From this observation I would conclude that the cracks investigated by Commonwealth Edison in 1979 were in fact the largest ones visible then also. I observed that at the end of the repaired area there had been no continuation or propagation of cracking since 1979 from the cracked and unrepaired area into the patch. I did observe a small (probably width of 0.005") crack which crossed the patch (generally at 90°) and continued about two feet on one side of the patch and about three feet on the other side of the patch. I attributed this to minor shrinkage that has occurred since 1979. Generally one can consider that 80-90% of the shrinkage takes place during the first year after placement and that this crack was a result of the later shrinkage.

The largest cracks I observed were on the order of 0.005" to 0.008" in width based on a wire feeler gauge I utilized. The maximum depth I was able to insert the probe was about 1/8". The cracks that were observed appeared to define no particular pattern with respect to embedded anchors, drilled-in anchors or the lines of distress that would develop as a result of excessive load or an understrength condition due to construction errors. Based on my observations I concluded that the roof does not show signs of distress as a result of cracking from any conditions related to external loads, drilling or construction errors. There are cracks, however, and these are to be expected in reinforced concrete construction. The shrinkage effects on the concrete in this particular roof framing system may be somewhat amplified due to differential shrinkage since the slab portions are relatively thin and can lose moisture fairly easy with the resulting shrinkage. The beam portions, on the other hand, are massive (3' x 4' in cross-section) and tend to have fewer losses and changes of moisture.

The embedments which were cast-in-place when the roof system concrete was placed consist of flat steel plates anchored by welded studs in the typical fashion. The condition of the concrete adjacent to these embedments showed some of the same minor cracks of from 0.005" to 0.008" in width. There appeared to be no consistency in the location of cracks to define a pattern that would relate to relatively heavily loaded anchors vs. lightly loaded anchors. In one instance an anchor judged visually to be relatively heavily loaded had no crack adjacent to the anchor, whereas an anchor plate with no load (unused) had some adjacent cracking. There was also no evidence to indicate that the unused anchor had ever been loaded.

The anchor bolts which were the drilled-in expansion type were used for attachments where the embedments could not serve as a result of there location or configuration. I observed several locations where drilling had apparently been started and was terminated as a result of apparently contacting reinforcing steel. Three specific anchors were examined in detail from the field observations back to the design layout and control of the design (anchors CC-13, CC-93 and CC-CP-7). All locations found where drilling was terminated due to contacting rebar were apparently patched as indicated by the licensee since no open holes were found. Several unused drilled-in anchors were observed and probably were left unused due to relocation of other anchors on a specific anchor plate with multiple anchor bolts.

It was stated by Mr. D. Shamblin of Commonwealth Edison that he knew of no core holes (cut all the way through) made in the roof slab. All through-slab penetrations were cast in place with sleeves or blocked out during concrete placement. I observed no indications of any core holes.

During the drilling operation in the off-gas filter building there were no cuts made through reinforcing steel. There were only hits or nicks made on the reinforcing steel as it was contacted. These hits were recorded when they occurred and they were illustrated on S&L Drawing, RHS-188, Rev. J. No observations could be made in the field but it is my opinion that these nicks will not have any significant effect on the off-gas filter building roof.

Conclusions

As a result of my review of the pertinent documents, discussions with cognizant individuals and my independent field observations and measurements I have concluded the following:

1. The off-gas filter building is a non-safety related bulding which contains equipment which has no function in preventing or mitigating accidents or accident conditions.

2. The reinforced concrete roof of the off-gas filter bulding has slight deviations of the thickness of the 12 inch thick slab as specified by the design drawings. The range of deviations of +1½" and -13/16" will have no significant effect on structural behavior.

3. The loading imposed by the temporary construction related transformer on the roof of the off-gas filter bulding did not exceed the design loads and did not cause structural distress.

4. The current condition of the roof of the off-gas filter building which includes existing cracks, embedments, anchor bolts and nicked reinforcing

steel is acceptable and there is every reason to expect that the roof system can fully carry the design live load of 100 psf while remaining in a serviceable condition in performing its intended function over its service life.

Attachments:
1. Request for Assistance
2. Section 3.2 and Table 3.2-1 of the SAR
3. Extracts from SER (NUREG 0519)
4. Extracts from Meeting Transcript (3/31/82)

Robert E. Shewmaker 4/7/82
Date

Robert E. Shewmaker, P.E.
Senior Civil-Structural Engineer
Office of Inspection and Enforcement

Attachment 1

UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION III
799 ROOSEVELT ROAD
GLEN ELLYN, ILLINOIS 60137

March 30, 1982

MEMORANDUM FOR: R. C. DeYoung, Director, Office of Inspection and Enforcement

FROM: James G. Keppler, Regional Administrator, Region III

SUBJECT: LA SALLE COUNTY NUCLEAR STATION - PETITION FROM ILLINOIS ATTORNEY GENERAL

As you know, on March 24, 1982, the Illinois Attorney General petitioned the NRC to suspend licensing proceedings at La Salle pending investigation of recent allegations and to institute a Show Cause Hearing with Illinois as a party to the Hearing. The allegations deal with the overall adequacy of safety related structures as a result of widespread rebar cutting and specific structural deficiencies in the roof of the off-gas building.

A conference call was held on March 29 involving Messrs. Denton, Case, Stello, DeYoung and Keppler to discuss the handling of these investigations. We agreed that, because the petition expresses concern that the off-gas building deficiencies had been verbally communicated earlier to NRC and that the NRC had concluded an investigation of these alleged deficiencies was not warranted, it would be prudent to have an independent review of this allegation by IE (since IE was not involved in the consideration not to investigate). This review should address both the technical adequacy of the off-gas building concerns as well as the NRC's handling of the earlier notification in this regard. With respect to the concerns associated with cutting through rebar this matter will be reviewed by Region III with technical assistance from NRR.

I realize your staff is already depleted as a result of other investigation assistance you are giving us, and your willingness to assist in this effort is genuinely appreciated.

James G. Keppler

James G. Keppler
Regional Administrator

cc: V. Stello, DEDROGR
H. R. Denton, NRR

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3.2 CLASSIFICATION OF STRUCTURES, COMPONENTS, AND SYSTEMS

Certain structures, components, and systems of the nuclear plant are considered important to nuclear safety because they perform safety actions required to prevent or mitigate the consequences of abnormal operational transients or accidents. The purpose of this section is to classify structures, components, and systems according to the importance of the safety function they perform. In addition, design requirements are placed upon such equipment to ensure the proper performance of safety actions when required.

3.2.1 Seismic Classification

Plant structures, systems, and components important to safety are designed to withstand the effects of a safe shutdown earthquake (SSE) and remain functional, if they are required to ensure:

a. the integrity of the reactor coolant pressure boundary,

b. the capability to shut down the reactor and maintain it in a safe condition, or

c. the capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures in excess of the guideline exposures of 10 CFR 100.

Plant structures, systems, and components, including their foundations and supports, designed to remain functional in the event of a SSE are designated as Seismic Category I, as indicated in Table 3.2-1.

All Seismic Category I structures, systems, and components are analyzed under the loading conditions of the SSE and operating-basis earthquake (OBE). Since the two earthquakes vary in intensity, the design of Seismic Category I structures, components, equipment, and systems to resist each earthquake and other loads are based on levels of material stress or load factors, whichever is applicable, and yield margins of safety appropriate for each earthquake. The margin of safety provided for such structures, components, equipment, and systems ensures that their design functions are not jeopardized. For further details of seismic design criteria refer to:

a. mechanical, in Subsection 3.7.3;

b. electrical, in Section 3.10;

c. structural, in Subsection 3.7.2; and

d. instrumentation and controls, in Section 3.10.

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3.2.2 System Quality Group Classifications

System quality group classifications have been determined for each water, steam, or radioactive waste-containing component of those applicable fluid systems which are relied upon to:

a. prevent or mitigate the consequences of accidents and malfunctions originating within the reactor coolant pressure boundary,

b. permit shutdown of the reactor and maintain it in the safe shutdown condition, and

c. contain radioactive material in large quantity or concentration.

A tabulation of quality group classification for each structure, system and component is shown in Table 3.2-1 under the heading, "Quality Group Classification." Figures 3.2-1 and 3.2-2 are diagrams which depict the relative locations of these structures, systems and components along with their quality group classification.

The implementation of the code requirements outlined in Tables 3.2-1, 3.2-2, 3.2-3, and 3.2-4 for fluid system components is discussed in Sections 3.7 and 3.9.

A boiling water reactor has a number of structures, systems, and components in the power conversion or other portions of the facility which have no direct safety function, but which may be connected to, or influenced by, the equipment within the nuclear safety-related classifications defined previously. Such structures, systems, and components are designated as "other."

The design requirements for equipment classified as "other" are specified by the designer with appropriate consideration of the intended service of the equipment and expected plant and environmental conditions under which it operates. Where possible, design requirements are based on applicable industry codes and standards. When these are not available, the designer relies on accepted industry or engineering practice.

Structures, systems, and components whose safety functions require conformance to the quality assurance requirement of 10 CFR 50, Appendix B, are summarized in Table 3.2-1 under the heading, "Quality Assurance Requirements." The quality assurance program is described in Chapter 17.0.

TABLE 3.2-1 (cont'd)

	PRINCIPAL COMPONENT(1)	LOCATION(3)	SEISMIC(5) CATEGORY	QUALITY (4a) GROUP CLASSIFICATION	QUALITY (4b) ASSURANCE REQUIREMENT	ELECTRICAL (4c) CLASSIFICATION	PURCHASE DATE(2)	COMMENTS
XXII.	**Local Panels**							
	1. Electrical panels with a safety function	A, RB	I	NA	I	1E	4-74	(15)
	2. Cable, with a safety function	A, RB	I	NA	I	1E	10-75	
	3. Remote shutdown panel	A	I	NA	I	1E	10-74	
XXIII.	**Off-Gas System (2)**							
	1. Atmospheric glycol tanks	F	II	D	II	NA	10-71	
	2. Heat exchangers	F, T	II	C	II	NA	10-74	
	3. Piping and valves (downstream of steam jet air ejectors)	T, F, O	II	C	II	NON 1E	9-74	
	4. Piping and valves (up to and including air ejector)	T	II	D	II	NON 1E	9-74	
	5. Valves	T, F	II	C	II	NON 1E		
	6. Steam jet air ejectors	T	II	D	II	NA	2-72	
	7. Charcoal vessels	F	II	C	II	NA	10-71	
	8. Recombiners	T	II	C	II	NA	10-71	
	9. Filters	F	II	C	II	NA	10-71	
	10. Afterfilter	F	II	C	II	NA	10-71	
	11. Reheater	F	II	--	II	NON 1E	1-72	
XXIV.	**Service Water System**							
	1. Piping	RB, O, L, A, T	II	D	II	NA	9-74	
	2. Strainers	L	II	D	II	NA	7-73	
	3. Pumps	L	II	D	II	NA	7-73	
	4. Pump motors	L	II	--	II	NON 1E	7-73	
	5. Valves	T, O, L, A, RB	II	D	II	NA	6-73	
	6. Electrical & instrument Modules	RB, L, A	II	--	II	NON 1E	--	
	7. Cable	RB, O, L, A, T	II	--	II	NON 1E	10-75	

T= turbine Bldg. Seismic Cat II
F= off-gas filter Bldg " "

TABLE 3.2-1 (Cont'd)

PRINCIPAL COMPONENT(1)	LOCATION(3)	SEISMIC(5) CATEGORY	QUALITY(4a) GROUP CLASSIFICATION	QUALITY(4b) ASSURANCE REQUIREMENT	ELECTRICAL(4c) CLASSIFICATION	PURCHASE DATE(2)	COMMENTS
XLII. Civil Structures							
1. Reactor building	RB	I	NA	I	NA		(22)
2. Lake screen house	L	NA	NA	II	NA		(22,27)
3. Radwaste building	RW	NA	NA	II	NA		(22)
4. Auxiliary building	A	I	NA	I	NA		(22)
5. Turbine building	T	NA	NA	II	NA		(22)
6. Off-gas filter building	F	II	NA	II	NA		(22)
7. Steam tunnel	A	I	NA	I	NA		(22)
8. River screen house	G	II	NA	II	NA		(22)
9. Diesel-generator building	RB	I	NA	I	NA		(22)
10. Auxiliary Spillway	O	NA	NA	II	NA		(22)
11. Cooling Lake Embankment	O	I	NA	II	NA		(22)
12. Submerged CSCS Pond (Ultimate Heat Sink)	O	I	NA	I	NA		(22)
13. Biological Shield	PC	I	NA	I	NA		(22)
14. Primary Containment	PC	I	NA	I	NA		(22)
a. Vacuum breaker piping	PC/RB	I	B	I	NA	9-74	
b. Vacuum breaker valves	PC/RB	I	B	I	1E		
c. Maintenance butterfly valves	PC/RB	I	B	I	NA		
d. Suppression vent downcomers	PC	I	NA	I	NA	----	
XLIII. MSIV Leakage Control System							
1. Piping, within RCPB isolation valves	RB	I	A	I	NA	9-74	
2. Piping, other upstream system lines	RB	I	B	I	NA	9-74	
3. Piping, downstream system from steamline connection to first valve	RB	I	D+	II	NA		
4. Piping, other downstream system lines	RB	I	B	I	NA	9-74	(7,8)
5. Valves, within RCPB	RB	I	A	I	1E	12-73	
6. Valves, other	RB	I	B	I	1E	12-73	
7. Heater	RB	I	NA	I	1E	5-76	
8. Blowers	RB	I	B	I	1E	11-75	
9. Electrical modules with a safety function	RB	I	NA	I	1E	--	
10. Cables, with a safety function	RB	I	NA	I	1E	10-75	

TABLE 3.2-1 (Cont'd)

EQUIPMENT CLASSIFICATION COMMENTS

(1) A module is an assembly of interconnected components which constitute an identifiable device or piece of equipment. For example, electrical modules include sensors (including electromechanical), power supplies, and signal processors; and mechanical modules include filters, strainers, and flow (element) assemblies/ orifices.

(2) Purchase order dates (month/year) are given for equipment as a basis for determining certain applicable codes on Tables 3.2-2, 3.2-3, and 3.2-4. Where two dates are given and indicated with a slash between them (e.g., 9-70/5-71) the first date corresponds to Unit 1 and the second date corresponds to Unit 2. Where two dates are given with a comma between (e.g., 9-70, 5-71), multiple purchase orders apply.

(3) PC = within primary containment
RB = within reactor building
O = outdoors onsite
L = lake screen house
A = auxiliary building
T = turbine building
RW = radwaste building
F = off-gas filter building
-- = all buildings except O, L

(4) a. Quality group classification per Tables 3.2-2, 3.2-3, and 3.2-4. Group "E" components are special engineered components in accordance with the codes and standards specified in the notes and comments for this Table.

b. I - The equipment meets the quality assurance requirements of 10 CFR 50, Appendix B.

II - The equipment is not required to meet the quality assurance requirements of 10 CFR 50, Appendix B.

c. 1E - Electrical equipment that meets the quality assurance standards of NRC guidelines and IEEE Standard 323-1971. Non-1E Electrical equipment that is not required to meet 1E requirements.
NA - not applicable because the equipment is not electrical.

TABLE 3.2-1 (Cont'd)

(5) I - The equipment is designed in accordance with the seismic requirements for the SSE.

II - The seismic requirements for the SSE are not applicable to the equipment.

(6) The control rod drive insert and withdraw lines from the drive flange up to and including the first valve on the hydraulic control unit are Quality Group B.

(7) The main steamlines between the outermost containment isolation valve up to the turbine stop valve, the main turbine bypass lines up to the turbine bypass valve, and all branch lines (2-1/2 inch nominal size and larger) connected to these portions of the main steam and turbine bypass lines up to the first valve capable of timely actuation are classified as D+. These sections of pipes meet all of the pressure integrity requirements of code practice for steam power plants plus the following additional requirements:

a. All longitudinal and circumferential butt weld joints are radiographed (or ultrasonically tested to equivalent standards). Where size or configuration does not permit effective volumetric examination, magnetic particle or liquid penetrant examination is substituted. Examination procedures and acceptance standards are at least equivalent to those specified as supplementary types of examination, in ANSI B31.1 Code.

b. All fillet and socket welds are examined by either magnetic particle or liquid penetrant methods. All structural attachment welds to pressure-retaining materials are examined by either magnetic particle or liquid penetrant methods. Examination procedures and acceptance standards are at least equivalent to those specified as supplementary types of examinations, in ANSI B31.1 Code.

c. All inspection records are maintained for the life of the plant. These records include data pertaining to qualification of inspection personnel, examination procedures, and examination results.

TABLE 3.2-1 (Cont'd)

(20) The unprocessed radwaste piping will meet Group D requirements and the following supplementary requirements:

a. Piping

For sizes over 4 inches nominal, random radiography of 20% of the joints was performed on girth and longitudinal butt-welds. Sockets and fillet welds in sizes over 4-inch nominal will be given random magnetic particle and liquid penetrant examination on 20% of the joints.

b. Pumps and valves

Welds in pumps and valves of pipe size over 4-inch was given random magnetic particle or liquid penetrant examination. Random examination is defined as examination of the linear dimension of a weld in a pump or valve with piping connecting over 10-inch nominal size or as examination of all of the welding in 20% of the pump and valves with piping connecting 10-inch nominal or less.

(21) Quality group classification requirements do not apply to piping and components supplied by the diesel engine manufacturer as an integral part of the diesel-generator unit. In this case, the manufacturer's standards are used with the intent that the piping or component is to function as reliably as possible.

(22) Civil structures were used in missile analyses as barriers. No individual missile barriers other than civil structures were credited.

(23) Includes Scram Discharge Volume Accumulators.

(24) Expendables and Consumables are purchased per original specification and stored under controlled conditions.

(25) Includes raceway installations containing Class 1E cables and other raceway installations required to meet Seismic Category I requirements (those whose failure during a seismic event may result in damage to any Class 1E or other safety-related system or component.

(26) Subsystems required for post-LOCA monitoring include containment hydrogen monitoring, containment pressure monitoring, containment temperature monitoring, suppression pool water level monitoring, suppression pool water temperature monitoring, and containment high-range radiation monitoring. Subsystems not required for post-LOCA monitoring

3 DESIGN CRITERIA FOR STRUCTURES, SYSTEMS AND COMPONENTS

3.1 Conformance with Nuclear Regulatory Commission General Design Criteria

In Section 3.0 of the Final Safety Analysis Report, the applicant presented an evaluation of the design bases for La Salle against the Commission's General Design Criteria listed in Appendix A of 10 CFR Part 50. We evaluated the final design and the design criteria and conclude, subject to the applicant's adoption of the additional requirements made by us as discussed in this report, that La Salle has been designed, can be constructed and can be operated to meet the requirements of the General Design Criteria.

3.1.1 Conformance with Industry Codes and Standards

Our review of structures, systems and components relies extensively on the application of industry codes and standards that have been used as accepted industry practice. These codes and standards, as cited in this report and the attached bibliography, have been previously reviewed and found acceptable by us; and have been incorporated into our Standard Review Plan.

3.2 Classification of Structures, Components, and Systems

3.2.1 Seismic Classification

Criterion 2 of the General Design Criteria requires that nuclear power plant structures, systems, and components important to safety be designed to withstand the effects of earthquakes without loss of capability to perform their safety function. These plant features are those necessary to assure (1) the integrity of the reactor coolant pressure boundary, (2) the capability to shut down the reactor and maintain it in a safe shutdown condition, or (3) the capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures comparable to 10 CFR Part 100 guideline exposures.

Structures, systems, and components important to safety that are required to be designed to withstand the effects of a safe shutdown earthquake and remain functional have in general been properly classified as seismic Category I items, in conformance with Regulatory Guide 1.29, "Seismic Design Classification," Revision 2 except for the following system. The applicant's nonseismic Category I classification of the cooling loop of the spent fuel pool cooling and cleanup system is not in conformance with Regulatory Guide 1.29.

As an alternate to a seismic Category I design cooling loop of the fuel pool cooling and cleanup system, the applicant has provided an analysis in the Final Safety Analysis Report that shows the radiological releases, following a postulated failure of this system to function, are a small fraction of the guideline values of 10 CPR Part 100. A seismic Category I cooling water makeup system to the pool is provided. For further review of the spent fuel pool cooling and cleanup system, see Section 9.1.3 of this report.

A18/1/3

All other structures, systems, and components that may be required for operation of the facility have been designed to nonseismic Category I requirements, including those portions of seismic Category I systems such as vent lines, fill lines, drain lines, and test lines on the downstream side of isolation valves which are not required to perform a safety function. Structures, systems, and components important to safety that have been designed to withstand the effects of a safe shutdown earthquake and remain functional are identified in an acceptable manner in Table 3.2-1 of the Final Safety Analysis Report. The basis for acceptance in our review has been conformance of the applicant's designs, design criteria, and design bases for structures, systems, and comopnents important to safety with the Commission's regulations as set forth in Criterion 2 of the General Design Criteria and to Regulatory Guide 1.29, our technical positions, and industry codes and standards.

Except for the cooling loop of the spent fuel pool cooling and cleanup system identified above, we conclude that structures, systems, and components important to safety that are designed to withstand the effects of a safety shutdown earthquake and remain functional have been properly classified as seismic Category I items in conformance with the Commission's regulations, the applicable regulatory guides, and industry codes and standards and are acceptable. Design of these items in accordance with seismic Category I requirements provides reasonable assurance that in the event of a safe shutdown earthquake, the plant will perform in a manner providing adequate safeguards to the health and safety of the public, and is acceptable.

3.2.2 System Quality Group Classification

Criterion 1 of the General Design Criteria requires the nuclear power plant systems and components important to safety shall be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety function to be performed. Fluid system pressure-retaining components important to safety have been designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety function to be performed. The applicant identified those fluid-containing components which are part of the reactor coolant pressure boundary and other fluid systems important to safety where reliance is placed on these systems: (1) to prevent or mitigate the consequences of accidents and malfunctions originating within the reactor coolant pressure boundary, (2) to permit shutdown of the reactor and maintain it in a safety shutdown condition, and (3) to contain radioactive material. These fluid systems have been classified in an acceptable manner in Table 3.2-1 of the Final Safety Analysis Report and on system piping and instrumentation diagrams in the Final Safety Analysis Report based on conformance with Regulatory Guide 1.26, "Quality Group Classification and Standards," Revision 3.

The applicant has applied Quality Groups A, B, C, and D in conformance to Regulatory Guide 1.26, to the fluid system pressure-retaining components important to safety. Those components that are classified Quality Group A, B, C, or D have been constructed to the codes and standards identified in Tables 3.2-2, 3.2-3, and 3.2-4 of the Final Safety Analysis Report.

The basis for acceptance in our review has been conformance of the applicant's designs, design criteria, and design bases for pressure-retaining components such as pressure vessels, heat exchangers, storage tanks, pumps, piping and

We determined that the liquid radwaste treatment system is capable of reducing the release of radioactive materials in liquid effluents to concentrations below the limits in 10 CFR Part 20, during periods of fission product leakage from the fuel at design levels.

Based on these findings, we conclude that the design of the liquid radioactive waste treatment systems is acceptable.

11.2.2 Gaseous Radioactive Waste Treatment System

The gaseous radioactive waste treatment system is designed to process gaseous wastes based on the origin of the wastes in the plant and the expected levels of radioactivity.

The gaseous waste treatment system consists of the main condenser offgas treatment system, mechanical vacuum pump offgas system, drywell purge system, gland seal condenser offgas system, and building ventilation systems.

The offgas treatment system, shared by Unit Nos. 1 and 2, is designed to collect and delay fission product noble gases removed from the condenser by the air ejectors. In the offgas treatment system, the gas from both units flows through a preheater, a recombiner, a condenser/separator, a 30-minute holdup pipe, a condenser/separator, a reheater, prefilter, eight charcoal beds in series, and an afterfilter. Except for the holdup pipe and the second reheater, the offgas treatment system consists of two separate trains of equipment which provide 100 percent redundancy in the processing of the gaseous wastes. The eight charcoal beds are designed to operate at 77 degrees Fahrenheit and 45 degrees Fahrenheit dew point and contain three tons of charcoal each. We consider the system capacity and design to be adequate for meeting the demands of the station during normal operation, including anticipated operational occurrences. The system design includes dual hydrogen analyzers upstream and downstream of the recombiner which will provide automatic dilution or activate an alarm upon exceeding a preset hydrogen concentration and indicate that switchover to the standby recombiner is required. In addition to the protective instrumentation, the offgas treatment system is designed to withstand a hydrogen explosion (design pressure, 350 pounds per square inch gauge). We find the design provisions incorporated to reduce the potential of hydrogen explosion and to mitigate the effects to be in accordance with the guidelines of Regulatory Guide 1.143 and are, therefore, acceptable.

The seismic and quality group classification of the offgas treatment system is based on criteria which were acceptable during the construction permit licensing stage, i.e., Quality Group C classification, nonseismic design for components. Although these criteria differ from the current criteria contained in Regulatory Guide 1.143, we have determined that the provisions incorporated in the design of the offgas treatment system are acceptable under the guidelines of Regulatory Guide 1.143. The process offgas system is located in the offgas filter building which is a nonseismic Category I structure. The charcoal vessels were designed to Quality Group C and meet American Society of Mechanical Engineers Code Section III, Class 3, 1971. The parameters of the principal components considered in the process offgas system evaluation are listed in Table 11.4. We find the process offgas system and the structure housing the system to be acceptable.

Attachment 4

NUCLEAR REGULATORY COMMISSION

ORIGINAL

In the Matter of: :

COMMONWEALTH EDISON COMPANY :

LaSalle County Nuclear Generating Station, Unit 1 and Unit 2 :

DOCKET NOS. 50-373 and 50-374

DATE: March 31, 1982 PAGES: 1 - 77

AT: Bethesda, Maryland

ALDERSON REPORTING

400 Virginia Ave., S.W. Washington, D. C. 20024

Telephone: (202) 554-2345

A18/1/4

MR. DELGEORGE: What I would like to do is review the allegations presented in the petition as we understand them, stating the facts and the information we have which we think will resolve the concerns that have been raised in your mind.

I would like to start with the questions raised relative to the off-gas building because we feel that to be a less complicated issue that can be more easily dispositioned.

First, there is an allegation that the roof thickness is eight inches as opposed to the 12 inch design thickness. I would like to say at the outset that although this building is a non-safety related building containing no safety-related equipment and not requiring the implementation of our quality assurance program, we did in fact apply our quality assurance program to the construction of this building, which has given us greater confidence in the accuracy of the information that we will be providing to you.

MR. DELGEORGE: I am ready. The last allegation suggested that the concrete associated with this slab had been cracked substantially. Commonwealth Edison discovered surface cracking of the subject slab through its own site quality assurance department in September 1979. As a result of the deficiency identified, an inquiry was made at that time which included an engineering evaluation and which also included the tracing of the crack depth by chipping at the concrete in the vicinity of the cracks.

As a result of that review, it was established that the crack depth did not exceed one quarter inch; that the cracking was, in fact, surface cracking, and as

a result, it was patched. We have no reason to believe, based on that investigation, that the cracking alleged is the result of drilling of anchor bolt holes. It is our opinion, based on that evaluation, that the cracks observed are normal shrinkage cracks associated with this type of slab.

MR. DENTON: Let me ask the project manager what categorization we gave that roof.

MR. BOURNIA: It is a non-safety grade building. I have the reviewer here. We did not consider this as a safety grade building.

MR. DENTON: What is under the roof?

MR. BOURNIA: What is this?

MR. DENTON: What is under it?

MR. DELGEORGE: That is described in our report. The concrete enclosure above-grade as a part of the off-gas roof is a non-safety related structure which houses off-gas building, heating/ventilating/and air conditioning, air handling units, HVAC, water cooled

condensing units, HVAC exhaust filter units, HVAC control panels and associated motor control centers and switchgear.

MR. DENTON: Does that mean there is no Category 1 safety-related equipment in that building?

MR. DELGEORGE: Yes, sir.

MR. DENTON: Any questions? We can come back to this, but I thought we would give the company a chance.

MR. NORELIUS: We received allegations on this some months ago and evaluated it in-office. I do not have those with me. I am not sure that I know they say exactly what she said, and I have not read them carefully. But we were aware of the allegation. It was evaluated within our office and I think, in recognition of our manpower considerations, we chose not to delve deeply into this at the field level because of its

Category 2 nature.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

April 17, 1982

PRINCIPAL STAFF			
DIR		FSIS	
D/D		CAO	
A/D		SLO	
DP&PI			
DF&TI			
DEP&OS		File	KOB

Docket Nos.: 50-373
and 50-374

Tyrone C. Fahner, Esquire
Attorney General
State of Illinois
160 North La Salle Street
Chicago, Illinois 60601

Dear Mr. Fahner:

This letter is to acknowledge receipt of your Request to Institute a Show Cause Proceeding and for Other Relief (Petition) dated March 24, 1982, filed with the Nuclear Regulatory Commission on behalf of the State of Illinois. The Petition principally seeks institution of a show cause proceeding under 10 C.F.R. § 2.202, and suspension of further consideration of the operating license application regarding the La Salle County Nuclear Generating Station, Units 1 and 2 of the Commonwealth Edison Company, based on certain allegedly newly discovered safety issues. The issues raised in your Petition are two in number. The first issue relates to the drilling of holes in the concrete walls, floors, and ceilings of certain buildings at the La Salle facility, including in some instances severance of steel reinforcing bar (rebar) with the potential for affecting structural integrity. The second issue relates to the structural adequacy of the off-gas building roof. Your Petition alleges substantial cracking of this roof and the possibility that the roof thickness does not meet design specifications.

Your Petition has been referred to me by the Commission for consideration pursuant to 10 C.F.R. § 2.206, and appropriate action will be taken on your petition within a reasonable time. The NRC Staff is investigating the allegations contained in your Petition 1/.

1/ Your Petition at page 2 states that there is an operating license proceeding presently before the Commission and that no hearing has been requested or noticed in this proceeding. A Notice of Receipt of Application for Facility Operating Licenses; Notice of Availability of Applicant's Environmental Report; and Notice of Consideration of Issuance of Facility Operating Licenses and Notice of Opportunity for Hearing, were published in the Federal Register regarding this facility on June 3, 1977. Consequently, a hearing opportunity regarding the La Salle facility operating licenses was noticed. No hearing was requested pursuant to the notice and consequently no operating license "proceeding" is before the Commission. I am presently considering Commonwealth Edison's license application, and have under consideration the issuance of a fuel load and low power testing (up to 5% rated power) license in the very near future.

A19 ~~8205060124~~ PDR/LPDR

APR 23 1982

As one step in our investigation, we contacted the Commonwealth Edison Company (CECO) on March 29, 1982, and asked that they meet with us on March 31, 1982, to present information with regard to the matters you raised. This meeting was held in Bethesda, Maryland on March 31, 1982, and Ms. Judith S. Goodie, Esq. attended the meeting as a representative of your office. A verbatim transcript was taken of that meeting and has been made available to you.

Our Region III office is determining whether the applicant has adequately implemented its rebar damage identification and assessment activity in accordance with suitable procedures during the construction process. Also, our Office of Inspection and Enforcement is reviewing Region III's handling of the allegation regarding the off-gas building roof, and is further determining whether the off-gas building roof meets its design requirements as stated by CECO at the March 31 meeting and in the FSAR for the La Salle facility. In conjunction with Region III activities, my office is assessing the technical adequacy of the applicant's engineering design assessment in the affected areas as a result of these allegations.

I have considered your request at Section III.1 of your Petition for an immediate suspension of consideration of Commonwealth Edison's application for a fuel load and low power testing license for La Salle Unit 1 until the allegations contained in your Petition are investigated and a decision made regarding institution of the requested show cause proceeding under 10 C.F.R. § 2.202. 2/

By letter dated February 4, 1982, CECO has addressed its startup test schedule for La Salle Unit 1. In this letter, the applicant estimates approximately 60 to 90 days will elapse from the time of issuance of the low power license until the time the reactor achieves initial criticality. The applicant's best current estimate is that this elapsed time will be approximately 61 days. During this time, there will be preliminary startup activities going on; however, since the reactor will not have been brought critical, there will be essentially no fission products in the core and no significant radiological hazard. Also, there will be no significant build-up of residual core activity through zero power physics testing.

During the zero power physics testing, the off-gas building would not receive any radioactive materials and thereby pose no public health and safety hazard should its roof fail. Also, continued structural integrity of concrete which

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2/ Your Petition at page 9 also seeks suspension or stay of all "proceedings concerning Edison's applications for operating licenses" for the La Salle facility. As discussed in footnote 1, *supra*, there are no such "proceedings" pending. The only relevant pending matter is my consideration of Commonwealth Edison's license application including its request for a fuel load and low power testing license.

may have been affected by the alleged drilling and boring is not essential for even, in the most severe incident which could be postulated, the radioactive releases, would be insubstantial. The low fission product inventory also answers the concern raised in your Petition at Section II.8 that fuel loading of Unit 1 should be postponed "until the Commission fully examines the potential safety hazard presented by the cutting of reinforcing steel as alleged herein" on the grounds that the presence of the fuel in the structure of Unit 1 will interfere with the investigation of the allegations and with any corrective measures that might be ordered. The insignificant build-up of residual core activity through zero power physics testing presents no significant impediment to completing the investigative efforts associated with the alleged concerns.

In our efforts to investigate these matters, members of the NRC staff have made visits to the plant and to the Architect/Engineer's offices, Sargent and Lundy; have reviewed documents including structural drawings and several calculations performed by Sargent & Lundy and have held discussions with key applicant and Sargent & Lundy personnel (see Enclosures 1 and 2). While the reports on these efforts have not yet been finalized, the allegations set forth in your petition have not been substantiated.

We expect to complete our ongoing investigation of the allegations raised by your Petition in the next 30 to 60 days and we will make our findings at that time. In the interim, because there is no significant threat to the health and safety of the public represented by a core that has not experienced operation beyond initial criticality and zero power testing, I have determined the immediate suspension of Commonwealth Edison's request for a fuel load and low power operating license to be unwarranted. Any such license, however, will be conditioned to require prior NRC staff approval for any power operation following initial criticality and zero power physics testing. Our approval will not be given unless warranted by the results of our current ongoing investigations.

The NRC Staff will continue to review the remaining matters raised in your Petition, and I will issue a decision with regard to them in the reasonably near future. I will also consider your Petition in any future licensing actions I take with respect to the La Salle facility.

I enclose for your information a copy of the Notice that is being filed for publication with the Office of the Federal Register.

Sincerely,

Harold R. Denton

Harold R. Denton, Director
Office of Nuclear Reactor Regulation

Enclosures:
As stated

cc: See next page

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Director of Nuclear Licensing
Commonwealth Edison Company
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Chairman
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Department of Public Health
ATTN: Chief, Division of Nuclear Safety
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Springfield, Illinois 62761

The Honorable Tom Corcoran
United States House of Representatives
Washington, D. C. 20515

Chairman
Illinois Commerce Commission
Leland Building
527 East Capitol Avenue
Springfield, Illinois 62706

ENCLOSURE 1

A19/1

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

APR 14 1982

MEMORANDUM FOR: Franz P. Schauer, Chief
Structural Engineering Branch
Division of Engineering

THRU: Dave Jeng, Section A Leader
Structural Engineering Branch
Division of Engineering

FROM: Romuald E. Lipinski and Sai P. Chan
Structural Engineering Branch
Division of Engineering

SUBJECT: TRIP REPORT - VISIT TO LA SALLE PLANT AND MEETING ON HOLE-DRILLING AND CUT REBARS IN CONCRETE

In accordance with your instruction, we visited the La Salle Plant at La Salle County, Illinois on April 8, 1982 and attended the meeting at Sargent and Lundy Engineers, Chicago, Illinois on April 9, 1982. The purpose of this trip is to familiarize ourselves with the general design of the plant and to gather sufficient information so as to fairly assess the applicant's practice of drilling holes through concrete elements in response to the allegation made by the Attorney General, State of Illinois.

At the plant site, we have identified and verified some groups of drilled holes in the containment and auxiliary buildings as indicated in the set of drawings that the applicant submitted at the March 31 presentation meeting at Bethesda. We have also witnessed measurement and verification of several dimensions to support the claim that the thickness of the roof slab of the off gas building is 12 inches.

On April 9, 1982, a meeting was held at the office of Sargent and Lundy Engineers (S & L), Chicago, Illinois to discuss the program of documentation of drilling and coring of holes and cutting rebars. The attendance list is attached.

We reviewed several calculations performed by the S & L by which the structural margins of the areas where drilling took place have been assessed. On the basis of the information gathered at the site and during the meeting at the S & L offices we established the following:

1. The roof of the off gas building is 12 inches thick, and

2. The controls and engineering evaluation of the effect of drillings were such that there is a reasonable assurance that they will not result in unacceptable degradation of structural elements.

8205060135

A19/1

In view of the above we are of the opinion that the allegation filed by the Attorney General of the State of Illinois is not justified.

R. E. Lipinski

R. E. Lipinski
Structural Engineering Branch
Division of Engineering

S. P. Chan

S. P. Chan
Structural Engineering Branch
Division of Engineering

Enclosure: Attendance List

cc: R. Vollmer
D. Eisenhut
J. Knight
R. Tedesco
A. Schwencer
D. Jeng
R. Shewmaker (I & E)
F. Hawkings (Reg. III)
A. Bournia

CONTACT: S. P. Chan, SEB, X29534

ENCLOSURE 2

A19/2

ENCLOSURE

ATTENDANCE LIST

APRIL 9, 1982

CHICAGO, ILLINOIS

NRC

F. Hawkins - Region III

R. Lipinski - NRR/SEB

S. Chan - NRR/SEB

CEC

M. J. Morris

L. O. Del George

C. W. Schroeder

S & L

T. Linglais

S. M. Kazmi

V. Reklaitis

K. T. Kostal

L. P. Dolder

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CSN

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

APR 14 1982

MEMORANDUM FOR: ~~Franz P. Schauer, Chief~~
Structural Engineering Branch
Division of Engineering

THRU: Dave Jeng, Section A Leader
Structural Engineering Branch
Division of Engineering

FROM: Romuald E. Lipinski and Sai P. Chan
Structural Engineering Branch
Division of Engineering

SUBJECT: TRIP REPORT - VISIT TO LA SALLE PLANT AND MEETING ON HOLE-DRILLING AND CUT REBARS IN CONCRETE

In accordance with your instruction, we visited the La Salle Plant at La Salle County, Illinois on April 8, 1982 and attended the meeting at Sargent and Lundy Engineers, Chicago, Illinois on April 9, 1982. The purpose of this trip is to familiarize ourselves with the general design of the plant and to gather sufficient information so as to fairly assess the applicant's practice of drilling holes through concrete elements in response to the allegation made by the Attorney General, State of Illinois.

At the plant site, we have identified and verified some groups of drilled holes in the containment and auxiliary buildings as indicated in the set of drawings that the applicant submitted at the March 31 presentation meeting at Bethesda. We have also witnessed measurement and verification of several dimensions to support the claim that the thickness of the roof slab of the off gas building is 12 inches.

On April 9, 1982, a meeting was held at the office of Sargent and Lundy Engineers (S & L), Chicago, Illinois to discuss the program of documentation of drilling and coring of holes and cutting rebars. The attendance list is attached.

We reviewed several calculations performed by the S & L by which the structural margins of the areas where drilling took place have been assessed. On the basis of the information gathered at the site and during the meeting at the S & L offices we established the following:

1. The roof of the off gas building is 12 inches thick, and

2. The controls and engineering evaluation of the effect of drillings were such that there is a reasonable assurance that they will not result in unacceptable degradation of structural elements.

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In view of the above we are of the opinion that the allegation filed by the Attorney General of the State of Illinois is not justified.

R. E. Lipinski

R. E. Lipinski
Structural Engineering Branch
Division of Engineering

S. P. Chan

S. P. Chan
Structural Engineering Branch
Division of Engineering

Enclosure: Attendance List

cc: R. Vollmer
D. Eisenhut
J. Knight
R. Tedesco
A. Schwencer
D. Jeng
R. Shewmaker (I & E)
F. Hawkings (Reg. III)
A. Bournia

CONTACT: S. P. Chan, SEB, X29534

F

ENCLOSURE

ATTENDANCE LIST

APRIL 9, 1982

CHICAGO, ILLINOIS

NRC

F. Hawkins - Region III
R. Lipinski - NRR/SEB
S. Chan - NRR/SEB

CEC

M. J. Morris
L. O. Del George
C. W. Schroeder

S & L

T. Linglais
S. M. Kazmi
V. Reklaitis
K. T. Kostal
L. P. Dolder

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NUCLEAR REGULATORY COMMISSION

[Docket Nos. 50-373; 50-374]

COMMONWEALTH EDISON COMPANY

(La Salle County Nuclear Generating Station, Unit 1 and Unit 2)

Request for Action Under 10 C.F.R. 2.206

Notice is hereby given that by its Request to Institute a Show Cause Proceeding and for Other Relief dated March 24, 1982 (Petition), the Attorney General for the State of Illinois requested that certain actions be taken by the Nuclear Regulatory Commission with respect to the La Salle County Units 1 and 2 of the Commonwealth Edison Company, in light of alleged newly discovered safety issues. Alleged safety issues concern the drilling of holes into structural concrete at the La Salle facility and also concern the structural adequacy of the off-gas building roof for that facility. The relief requested included institution of a show cause proceeding pursuant to 10 C.F.R. § 2.202 and immediate suspension of further consideration of Edison's application for operating licenses. This request is being treated as a petition pursuant to 10 C.F.R. 2.206 of the Commission's regulations, and accordingly, action will be taken on the Petition within a reasonable time.

Copies of the Petition are available for inspection in the Commission's public document room at 1717 H Street, N.W., Washington,

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D.C. 20555, and in the local public document room for the La Salle County Plant, Units 1 and 2 located at Illinois Valley Community College, Rural Route #1, Oglesby, Illinois 61348.

Dated at Bethesda, Maryland this 17 th day of April, 1982.

FOR THE NUCLEAR REGULATORY COMMISSION

Harold R. Denton

Harold R. Denton
Director
Office of Nuclear Reactor Regulation

Mary. Mike
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and files

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

Docket Nos. 50-373

APR 1 7 1982

Mr. Cordell Reed
Vice President, Nuclear Operations
Commonwealth Edison Company
Post Office Box 767
Chicago, Illinois 60690

Dear Mr. Reed:

Subject: La Salle County Station, Unit 1 - Issuance of Facility Operating License

The U.S. Nuclear Regulatory Commission (NRC) has issued the enclosed Facility Operating License No. NPF-11 to Commonwealth Edison Company for La Salle County Station, Unit 1, located in Brookfield Township, La Salle County, Illinois. License No. NPF-11 authorizes operation of the La Salle County Station, Unit 1, at five percent power (166 megawatts thermal). Authorization to operate beyond five percent is still under consideration by the NRC. The issuance of this license authorizing operation at five percent of full power is without prejudice to future consideration by the Commission with respect to operation at power levels in excess of five percent.

Also enclosed is a copy of a related Federal Register notice which has been forwarded to the Office of the Federal Register for publication.

Two signed copies of Amendment No. 3 to Indemnity Agreement No. B-84 which covers the activities authorized under License No. NPF-11 are also enclosed. Please sign and return one copy to this office.

Supplement No. 3 to the Safety Evaluation Report for La Salle County Station, Units 1 and 2 has been issued. Two copies are enclosed; 18 additional copies will be forwarded to you in about a week.

Darrell G. Eisenhut, Director
Division of Licensing
Office of Nuclear Reactor Regulation

Enclosures:
1. Facility Operation License No. NPF-11
2. Federal Register Notice
3. Amendment No. 3 to Indemnity Agreement No. B-84
4. Supplement No. 3 to the SER (2)

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cc w/enclosures:
See next page

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PDR/LPDR

Mr. Louis O. DelGeorge
Director of Nuclear Licensing
Commonwealth Edison Company
P. O. Box 767
Chicago, Illinois 60690

cc: Philip P. Steptoe, Esquire
Suite 4200
One First National Plaza
Chicago, Illinois 60603

Dean Hansell, Esquire
Assistant Attorney General
188 West Randolph Street
Suite 2315
Chicago, Illinois 60601

~~Roger Walker,~~ Resident Inspector
LaSalle NPS, U.S.N.R.C.
P. O. Box 224
Marseilles, Illinois 61364

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

COMMONWEALTH EDISON COMPANY

DOCKET NO. 50-373

LA SALLE COUNTY STATION, UNIT 1

FACILITY OPERATING LICENSE

License No. NPF-11

1. The Nuclear Regulatory Commission (the Commission or the NRC) having found that:

   A. The application for a license filed by the Commonwealth Edison Company complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's regulations set forth in 10 CFR Chapter I, and all required notifications to other agencies or bodies have been duly made;

   B. Construction of the La Salle County Station, Unit 1 (the facility), has been substantially completed in conformity with Construction Permit No. CPPR-99 and the application, as amended, the provisions of the Act, and the regulations of the Commission;

   C. The facility will operate in conformity with the application, as amended, the provisions of the Act, and the regulations of the Commission;

   D. There is reasonable assurance: (i) that the activities authorized by this operating license can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations set forth in 10 CFR Chapter I;

   E. The Commonwealth Edison Company is technically qualified to engage in the activities authorized by this operating license in accordance with the Commission's regulations set forth in 10 CFR Chapter I;

   F. The Commonwealth Edison Company has satisfied the applicable provisions of 10 CFR Part 140, "Financial Protection Requirements and Indemnity Agreements," of the Commission's regulations;

   G. The issuance of this license will not be inimical to the common defense and security or to the health and safety of the public;

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H. After weighing the environmental, economic, technical, and other benefits of the facility against environmental and other costs and considering available alternatives, the issuance of Facility Operating License No. NPF-11, subject to the conditions for protection of the environment set forth herein, is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied; and

I. The receipt, possession, and use of source, by-product and special nuclear material as authorized by this license will be in accordance with the Commission's regulations in 10 CFR Parts 30, 40 and 70.

2. Based on the foregoing findings regarding this facility, Facility Operating License NPF-11 is hereby issued to the Commonwealth Edison Company (the licensee) to read as follows:

A. This license applies to the La Salle County Station, Unit 1, a boiling water nuclear reactor and associated equipment, owned by the Commonwealth Edison Company. The facility is located in Brookfield Township, La Salle County, Illinois, and is described in the licensee's "Final Safety Analysis Report," as supplemented and amended, and in the licensee's Environmental Report, as supplemented and amended.

B. Subject to the conditions and requirements incorporated herein, the Commission hereby licenses:

(1) Commonwealth Edison Company, pursuant to Section 103 of the Act and 10 CFR Part 50, "Licensing of Production and Utilization Facilities," to possess, use, and operate the facility at the designated location in Brookfield Township, La Salle County, Illinois, in accordance with the procedures and limitations set forth in this license;

(2) Commonwealth Edison Company, pursuant to the Act and 10 CFR Part 70, to receive, possess and use at any time special nuclear material as reactor fuel, in accordance with the limitations for storage and amounts required for reactor operation, as described in the Final Safety Analysis Report, as supplemented and amended;

(3) Commonwealth Edison Company, pursuant to the Act and 10 CFR Parts 30, 40, and 70, to receive, possess, and use at any time any byproduct, source and special nuclear material as sealed neutron sources for reactor startup, sealed sources for reactor instrumentation and radiation monitoring equipment calibration, and as fission detectors in amounts as required;

(4) Commonwealth Edison Company, pursuant to the Act and 10 CFR Parts 30, 40, and 70, to receive, possess, and use in amounts as required any byproduct, source or special nuclear material without restriction to chemical or physical form, for sample analysis or instrument calibration or associated with radioactive apparatus or components; and

(5) Commonwealth Edison Company, pursuant to the Act and 10 CFR Parts 30, 40 and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of the facility.

C. This license shall be deemed to contain and is subject to the conditions specified in the Commission's regulations set forth in 10 CFR Chapter I and is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

(1) Maximum Power Level

The licensee is authorized to operate the facility at reactor core power levels not in excess of 5 percent of full power (166 megawatts thermal) in accordance with the conditions specified herein and in Attachment 1 to this license. The preoperational tests, startup tests and other items identified in Attachment 1 to this license shall be completed as specified. Attachment 1 is an integral part of this license.

(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A and the Environmental Protection Plan contained in Appendix B attached hereto are hereby incorporated in this license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

(3) Conduct of Work Activities During Fuel Load and Initial Startup

The licensee shall review by committee all Unit 1 Preoperational Testing and System Demonstration activities performed concurrently with Unit 1 initial fuel loading or with the Unit 1 Startup Test Program to assure that the activity will not affect the safe performance of the Unit 1 fuel loading or the portion of the Unit 1 Startup Program being performed. The review shall address, as a minimum, system interaction, span of control, staffing, security and health physics, with respect to performance of the activity concurrently with the Unit 1 fuel loading or the portion of the Unit 1 Startup Program being performed. The committee for the review shall be composed of a least three members, knowledgeable in the above areas, and who meet the qualifications for professional-technical personnel specified by

section 4.4 of ANSI N18.7-1971. At least one of these three shall be a senior member of the Assistant Superintendent of Operation's staff.

(4) Resolution of Rebar Damage and Adequacy of Off-gas Building Roof

The licensee shall complete its assessment of the rebar damaged due to drilling and coring in concrete and the structural adequacy of the off-gas building roof. The results shall be reported to the NRC staff for review and approval, prior to power operation following initial criticality and zero power physics testing.

(5) Snubbers

(a) Prior to criticality, the licensee shall submit for NRC approval, a revised list of safety-related snubbers to be contained in Table 3.7.9-1 of the Technical Specifications to include such snubbers on lines 3 inches in diameter or less.

(b) Prior to startup after the first refueling outage, the licensee shall provide, as necessary, a revision to the Technical Specifications to remove snubbers that are determined to be unnecessary and replace them with rigid strut and rod assemblies.

(6) Deferred Preoperational Deficiencies

The licensee shall satisfactorily resolve those deficiencies which were deferred from the preoperational testing program on a schedule that shall assure that the capability of a system required to be operable by Technical Specification is not degraded.

(7) Surveillance of Tendons (Section 3.8.1*, SSER #3)

Prior to full power, the licensee shall supply the predicted lift-off forces required to complete Tables 4.6.1.5-1 and 4.6.1.5-2 of the Technical Specifications.

(8) Masonry Wall (Section 3.8.3, SER ,SSER #2)

Based on the findings of our preliminary review of the licensee's submittals and its commitments related to masonry wall evaluation, the following actions are required by the licensee:

(a) The present fixes for modifications implemented shall not preclude the option of implementing additional modifications if directed by our future review of the licensee's design criteria.

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*The parenthetical notation following the title of many license conditions denotes the section of the Safety Evaluation Report and/or its supplements wherein the license condition is discussed.

(b) Prior to startup after the first refueling outage, the licensee shall resolve the differences between our interim criteria and the criteria used by the licensee to the satisfaction of the staff and shall implement the required wall fixes or modifications that might result from such a resolution.

(9) Inservice Testing of Pumps and Valves (Section 3.9.6, SER)

Pursuant to 10 CFR Part 50.55a, the relief that the licensee has requested from the pump and valve testing requirements of 10 CFR Part 50, Section 55.55 (g)(2) and (g)(4)(i) is granted for that portion of the initial 120-month period during which we complete our review.

(10) Dynamic Qualification (3.10, SER, SSER #1, SSER #2)

(a) Prior to startup after the first refueling outage, the licensee shall complete any modifications or replacement of equipment as a result of the fatigue evaluation. ~~In the interim, the licensee shall document the occurrence of every safety relief valve actuation into the suppression pool, the associated cumulative damage factors calculated for typical representative equipment and kept up-to-date, and report to NRC any malfunction of equipment that occurs due to any safety relief discharge.~~

(b) Prior to startup after the first refueling outage, the licensee shall replace or modify the NSSS equipment (intermediate range monitor, C51-K-601A/H and two-inch air-operated globe valve, C11-F011) if the results of the requalification tests indicate either change is required.

(11) Environmental Qualifications (Section 3.11, SER, SSER #1, SSER #2)

(a) No later than June 30, 1982, the licensee shall be in compliance with the provisions of NUREG-0588, "Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment", for safety-related electrical equipment exposed to a harsh environment.

(b) Complete and auditable records must be available and maintained at a central location which describe the environmental qualification methods used for all safety-related electrical equipment in sufficient detail to document the degree of compliance with NUREG-0588. Such records shall be updated and maintained current as equipment is replaced, further tested, or otherwise further qualified to document complete compliance no later than June 30, 1982.

(c) The licensee shall complete the corrective actions stipulated in Appendix C to Supplement No. 2 of the Safety Evaluation Report by June 30, 1982.

(12) Seismic and Loss-of-Coolant Accident Loads (Section 4.2.3.4, SER, SSER #1, SSER #2)

(a) By July 30, 1982, the licensee shall submit to NRC a complete description of the analytical methods along with all analytical results necessary to show that La Salle fully meets the criteria of Appendix A to the Standard Review Plan, Section 4.2 (NUREG-0800) with regard to fuel assembly liftoff.

(b) Prior to startup after the first refueling outage, the review of the fuel assembly liftoff issue must be satisfactorily resolved to the satisfaction of the staff.

(13) Surveillance of Control Blade (Section 4.2.3.14, SER)

IE Bulletin No. 79-26, Revision 1, "Boron Loss from BWR Control Blades," describes certain actions to be taken by licensees to determine boron loss from BWR control blades. The licensee shall comply with items 1, 2 and 3 of this bulletin and submit a written response on item 3 within 30 days after plant startup following the first refueling outage.

(14) Scram Discharge Volume (Sections 4.6.2, SER and 6.3.2.3, SSER #2)

(a) Prior to startup after the first refueling outage, the licensee shall incorporate the following additional modifications into the scram discharge volume system:

(i) Redundant vent and drain valves, and

(ii) Diverse and redundant scram instrumentation for each instrumented volume, including both delta pressure sensors and float sensors.

(b) Prior to startup after the first refueling outage, the licensee shall complete system or procedural modifications, if required, as a result of the staff's completion of its review of the licensee's response to NUREG-0803.

(15) Low Pressure in Pump Discharge of the Control Rod Drive (Section 4.6.2, SSER #2)

Prior to startup after the first refueling outage, the licensee shall install instrumentation for an automatic scram that would shut down the reactor in the event of low control rod drive pump discharge pressure to be activated during startup and refueling modes only.

(16) Containment Long Term Program Load Specifications (6.2.1.1, SSER #2)

Prior to October 1, 1982, the licensee shall submit its confirmatory assessment of the containment design adequacy for pool dynamic loads (chugging, vent lateral and diaphragm reverse pressure) developed in conjuction with the Long Term Program and reported in NUREG-0808.

(17) Pressure Interlocks on Valves Interfacing at Low and High Pressure (Section 6.3.4, SSER #2)

Prior to startup after the first refueling outage, the licensee shall implement isolation protection in conformance to the requirements of Section 6.3 of the Standard Review Plan against overpressurization of the low pressure emergency core cooling systems (RHR/LPCI and LPCS) at the high and low pressure interface containing a check valve and a closed motor-operated valve.

(18) Compliance with Regulatory Guide 1.97 (Sections 7.5.2, SER)

By July 1, 1982, the licensee shall provide a plan for implementing modifications necessary to comply with Revision 2 of Regulatory Guide 1.97, "Instrumentation for Light Water Cooled Nuclear Power Plants to Assess Plant Conditions During and Following an Accident," dated December 1980.

(19) Additional Instrumentation and Control Concerns (Section 7.7.3.4, SSER #1)

The licensee shall resolve the following concerns to the NRC staff's satisfaction prior to startup after the first refueling outage:

(a) whether common electrical power sources or sensor malfunctions may cause multiple control systems failures, and

(b) whether high energy line breaks will result in unacceptable consequential control system failures.

(20) Low and/or Degraded Grid Voltage (Section 8.2.2.2, SER)

The licensee shall install a second level of undervoltage protection prior to startup after the first refueling outage.

(21) Reliability of Diesel-Generators (Sections 8.3.1.1, SER and 9.6.3.4, SER)

Prior to startup after the first refueling outage, the licensee shall implement the following design modifications with respect to diesel-generator reliability:

(a) A heavy duty turbocharger gear drive assembly be installed on the diesel-generators.

(b) A prelube pump, powered from a reliable direct current power supply, be installed in the system to operate in parallel with the engine-driven lube oil pump, or an alternative acceptable to the NRC shall be installed to preclude dry-starting of the diesel-engine.

(c) Controls and monitoring instrumentation be removed from the engine and engine skid, except instruments qualified for this location. The non-qualified control and monitoring instruments shall be installed on a free standing floor mounted panel and located on a vibration free floor area. If the floor is not vibration free, the panel shall be equipped with vibration mounts.

(22) Direct Current Power Systems (Section 8.3.1.2, SER)

Prior to startup after the first refueling outage for the 125 and 250-volt direct current systems for Divisions 1 and 2 and the 125-volt Division 3 direct current system, the following additional instrumentation shall be provided in the control room: (1) Battery current (ammeter-charge/discharge), (2) Battery charger output voltage (voltmeter), (3) Battery charger output current (ammeter), (4) Battery high discharge rate alarm, and (5) Battery charger trouble alarm. ~~In the interim, the licensee shall implement approved procedures to monitor battery current, battery charger output voltage, and battery charger output current at the local panels at least once per eight hour shift.~~

(23) Reactor Containment Electrical Penetrations (Section 8.4.1, SER)

Prior to startup after the first refueling outage, a redundant fault current device (circuit breakers or fuses) shall be provided on each penetrating circuit that would limit a fault current surge to be less than the surge for which the penetration is qualified except for low energy (milliamps) instrument systems.

(24) Separation of Class 1E and Non-Class IE Cable Trays (Section 8.4.6.1, SER, SSER #1, SSER #2)

Prior to startup after the first refueling outage, the licensee shall provide adequate separation or barriers between Class 1E and adjacent non-Class 1E cable trays.

(25) Fire Protection Program (Section 9.5, SER, SSER #2, SSER #3)

(a) The licensee shall maintain in effect and fully implement all provisions of the approved fire protection plan. In addition, the licensee shall maintain the fire protection program set forth in Appendix R to 10 CFR Part 50, except for the following deviations:

(i) Hydrostatic hose tests in accordance with NEPA 1962-1979, and

(ii) No automatic fire detection systems in areas 2K/3K and 5B4.

(b) ~~Prior to initial criticality, the licensee shall install a 1-hour rated barrier on all four sides of a partially protected power cable pan and a general sprinkler system, both located in the diesel generator corridor.~~

(c) Prior to startup after the first refueling outage, the licensee shall provide fire protection systems in fire areas 2C/3C, 4C3 and 6E.

(d) Prior to startup after the first refueling outage, the licensee with respect to fire doors shall implement one of the following:

(i) 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 rating, or

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

(iii) Replace manufacturer's labeled doors and door frames with UL rated items.

(e) Prior to startup after the first refueling outage, the licensee shall demonstrate the adequacy of its fire protection for record storage.

(26) Radiation/Chemistry Technicians on the Backshift (Section 13.1, SSER#2)

(a) All Radiation/Chemistry Technicians on the backshift shall be trained per the La Salle's Traninging Qualification Guide. All such Technicians shall also have satisfactorily completed the following emergency response training:

(i) Tasks to be performed during the first 60 minutes of a serious emergency on the backshift;

(ii) Post-accident sampling and analysis for the first three hours of an emergency;

(iii) In-plant radiation surveys during an accident;

(iv) Use and interpretation of both portable and fixed area radiation monitoring equipment, such as the Eberline PING-3 and SAM-2;

(v) Interpretation of critical effluent monitoring data for assisting the Chief Engineer during the first hour of an accident (i.e., station vent monitor and standby gas treatment monitor);

(vi) First aid and bioassay techniques; and

(vii) Use of respiratory equipment during emergency situations.

(b) By June 1, 1983, the licensee shall have Radiation/Chemistry Technicians onshift for 24 hours per day who meet ANSI N18.1-1971 or who are qualified in accordance with a NRC approved alternative program.

(27) Industrial Security (Section 13.6, SER, SSER#3)

The licensee shall maintain in effect and fully implement all provisions of the Commission's approved physical security plan, guard training and qualification plan, and contingency plan, including amendments made pursuant to the authority of 10 CFR 50.54(p). The approved plans which contain safeguard information are collectively entitled: "La Salle County Station Security Plan Units 1 and 2," Revision 11, dated December 24, 1981; "La Salle County Station Guard Training and Qualification Plan," submitted by their letter dated August 16, 1979, as revised in August, 1980; and "La Salle Nuclear Power Station Contingency Plan, dated March, 1980, as revised by pages dated June, 1980.

The licensee is exempt from the commitment to fully implement those portions of the Security Plan as described in Items 1 and 2 in the licensee's letter dated April 1, 1982, provided that the compensatory measures delineated in the above referenced letter are in place. Compensatory measures as described in Item 3 in the April 1, 1982 letter are approved with full implementation of the security plan commitments to be accomplished no later than July 1, 1982.

The licensee is exempted from the provisions of 10 CFR 73.55(d)(9), but shall meet all other commitments of the physical security plan and the following additional items.

(a) Change all keys, locks, and combinations and related equipment used to control access to protected areas and vital areas at least every 12 months.

(b) Issue keys, locks, combinations, and other access control devices to protected and vital areas only to those individuals who possess access authorization to those areas.

(c) Change keys, locks, combinations, and related equipment to which an individual had access within 5 days and immediately for card keys after access authorization is withdrawn due to lack of trustworthiness, reliability, or inadequate work performance.

(28) <u>Initial Test Program (Section 14, SER)</u>

The licensee shall conduct the post-fuel-loading initial test program (set forth in Section 14 of the licensee's Final Safety Analysis Report, as amended) without making any major modifications of this program unless modifications have been identified and have received prior NRC approval. Major modifications are defined as:

(a) Elimination of any test identified in Section 14 of the licensee's Final Safety Analysis Report, as amended as being essential;

(b) Modification of test objectives, methods or acceptance criteria for any test identified in Section 14 of the licensee's Final Safety Analysis Report, as amended as being essential;

(c) Performance of any test at a power level different from that described in the program; and

(d) Failure to complete any tests included in the described program (planned or scheduled for power levels up to the authorized power level).

(29) <u>Assurance of Proper Design and Construction (Section 17.4, SSER #2)</u>

Prior to exceeding 5 percent of full power, the licensee shall have conducted an independent review of the mechanical and structural design of the loop C residual heat removal system, excluding all branch piping less than 3 inches, in the functioning mode of the low pressure injection system using loads resulting from

the actuation of the automatic depressurization system in conjunction with the operating basis earthquake to verify that this system has been designed and constructed in accordance with all pertinent NRC requirements. This verification review shall consider design, installation, inspection, testing, and any other aspects necessary to ensure conformance with the design. This review shall be performed independently of the licensee and its contractors who performed design and construction activities for the La Salle County Station, and it shall be completed to the satisfaction of NRC.

(30) NUREG-0737 Conditions (Section 22.2)

~~The licensee shall complete the following conditions to the satisfaction of the NRC. These conditions reference the appropriate items in Section 22.2, "TMI Action Plan Requirements for Applicants for Operating Licenses," in the Safety Evaluation Report and Supplements 1, 2 and 3, NUREG-0519.~~

(a) Shift Technical Advisor (I.A.1.1, SER, SSER #2)

The Shift Technical Advisor (STA) function shall be fulfilled by the Station Control Room Engineer (SCRE) who will be a designated SRO. However, if a SCRE is not available, the licensee shall provide a fully-trained on-shift technical advisor to the shift engineer (shift supervisor).

~~(b)~~ Nuclear Steam Supply System Vendor Review of Procedures (I.C.7, SER)

Prior to beginning low-power testing, the licensee must assure that the General Electric review of the power-ascension test procedures has been completed and the General Electric recommendations have been incorporated.

(c) Independent Safety Engineering Group (I.B.1.2, SER)

The licensee shall have an on-site independent engineering group.

~~(d)~~ Control Room Design Review (I.D.1, SER, SSER #2)

The licensee shall correct the design deficiencies identified in Appendix C of Supplement No. 1 to the Safety Evaluation Report, NUREG-0519 on the schedule prescribed therein.

(e) Training During Low-Power Testing (I.G.1, SER, SSER #2)

At least 4 weeks prior to performing the Special Test, Simulated Loss of Onsite and Offsite Alternating-Current Power Test, the licensee shall provide a safety analysis for the test and its procedures to NRC for review and approval.

~~(f)~~ Post Accident Sampling (II.B.3, SSER #2)

~~Prior to criticality, the licensee shall install and test a high radiation sampling system for obtaining reactor coolant and containment atmosphere sampling under degraded core accident conditions without excessive exposure.~~

~~(g)~~ Direct Indication of Safety/Relief Valve Position (II.D.3, SER, SSER #2)

Prior to startup after the first refueling outage, the licensee shall replace the safety/relief valve position indicator to a model that meets the IEEE Standards 323-1974 and 344-1975.

~~(h)~~ Additional Accident-Monitoring Instrumentation (II.F.1, SER, SSER #2)

Attachment 1, Noble Gas Effluent Monitor

~~Prior to criticality, the licensee shall install and have procedures approved by the NRC for noble gas effluent monitoring system at plant effluent pathways.~~

Attachment 2, Sampling and Analysis of Plant Effluents

~~Prior to criticality, the licensee shall install and have procedures approved by the NRC for radioiodine and particulate sampling and analysis system at plant effluent pathways.~~

~~(i)~~ Instrumentation for Detection of Inadequate Core Coolng (II.F.2, SER SSER #1, SSER #2)

By July 31, 1982, the licensee shall submit a report addressing the analysis performed by the BWR Owners Group regarding additional instrumentation relative to inadequate core cooling and that the licensee shall implement the staff's requirements after the completion of the staff's review of this report.

(j) Proper Functioning of Heat Removal Systems (II.K.1.22, SER, SSER #2, and II.K.3.13, SER, SSER #2)

The licensee shall implement the logic to restart automatically the core isolation cooling system prior to startup after the first refueling outage.

(k) Modify Break Detection Logic to Prevent Spurious Isolation of High Pressure Coolant Injection and Reactor Core Isolation Cooling System (II.K.3.15, SER, SSER #2)

Prior to startup after the first refueling outage, the licensee shall implement a circuit modification to assure that transients monitored by pressure instruments to sense flow in these two systems actually sense continuous high steam flow.

(l) Modification of Automatic Depressurization System Logic - Feasibility for Increased Diversity for Some Event Sequences (II.K.3.18, SER, SER #1, SSER #3)

(a) By October 1, 1982, the licensee shall evaluate the alternative design modifications of the BWR Owners Group relative to the logic for the automatic depressurization system, submit such evaluation, and propose modification to NRC for review and approval.

(b) Prior to startup after the first refueling outage, the licensee shall implement the approved alternative logic modification of the automatic depressurization system.

(m) Restart of Core Spray and Low Pressure Core Injection System (II.K.3.21, SER, SSER #2)

Prior to startup after the first refueling outage, the licensee shall provide an auto start for the high pressure core spray.

(n) Automatic Switchover of Reactor Core Isolation Cooling System Suction--Verify Procedures and Modify Design (II.K.3.22, SER)

Prior to startup after the first refueling outage, the licensee shall implement the automatic switchover of the reactor core isolation cooling system suction from the condensate storage tank to the suppression pool when the condensate storage tank level is low.

(8) Upgrade Emergency Support Facilities (III.A.1.2, SER, SSER #1)

The licensee shall complete its Emergency Response Facilities as follows:

(i)	Safety Parameter Display System	October 1, 1982
(ii)	Emergency Operations Facility	October 1,1982
(iii)	Technical Support Center	October 1, 1982

(9) Improving Licensee's Emergency Preparedness - Long Term (III.A.2, SER, SSER #1, SSER #2)

(1) Prior to exceeding five percent power, the licensee shall complete a successful emergency exercise with the La Salle facility and La Salle County.

(2) Prior to exceeding five percent power, a test shall be performed to demonstrate an adequate alerting/notification system.

(3) Prior to exceeding five percent power, the licensee shall demonstrate that the state of offsite preparedness provides assurance that adequate protective measures can and will be taken in the event of a radiological emergency. The use of 10 CFR 50.54(s)(2) to specify a period within which corrective actions must be taken to assure an adequate state of emergency preparedness will include instances where NRC finds that the lack of progress in completion of the procedures in the Federal Emergency Management Agency's proposed rule set forth in 44 CFR Part 350 is an indication that major substantive problems exist in achieving or maintaining an adequate state of preparedness. Any corrective period specified will relate to substantive problems identified by the Federal Emergency Management Agency.

(4) The licensee shall provide the interim meteorological improvement and shall provide the mechanism for long-term improvements as follows:

(i) Prior to exceeding five percent power, the licensee shall install a process computer with the capability to retrieve meteorological information that provides a redundant means for data access.

(ii) Prior to exceeding five percent power, the licensee shall propose a plan for meeting the meteorological and dose assessment capability guidance of Appendix 2, NUREG-0654, Revision 1 as follows:

installation of hardware and software capability described above by July 1, 1982; and

full operational capability described above by January 1, 1983.

(iii) Prior to exceeding five percent power, the licensee shall include a description of the dose calculational methodology with a Class A transport and diffusion module, and a description of an acceptable meteorological measurement preventative and corrective maintenance program in the radiological emergency plan.

D. Exemptions from certain requirements of Appendices G, H, J, R and §50.55(a) to 10 CFR Part 50 and 10 CFR Part 73 are described in the Safety Evaluation Report and Supplement No. 1, No. 2 and No. 3 to the Safety Evaluation Report. In addition, an exemption was requested until the completion of the first refueling from the requirements of 10 CFR §70.24 and an exemption from 10 CFR Part 50, Appendix E from performing a full scale exercise within one year before issuance of an operating license, both exemptions are described in Supplement No. 2 of the Safety Evaluation Report. These exemptions are authorized by law and will not endanger life or property or the common defense and security and are otherwise in the public interest. Therefore, these exemptions are hereby granted. The facility will operate, to the extent authorized herein, in conformity with the application, as amended, the provisions of the Act, and the rules and regulations of the Commission.

E. This license is subject to the following additional condition for the protection of the environment:

Before engaging in additional construction or operational activities which may result in a significant adverse environmental impact that was not evaluated or that is significantly greater than that evaluated in the Final Environmental Statement and its Addendum, the licensee shall provide a written notification to the Director of the Office of Nuclear Reactor Regulation and receive written approval from that office before proceeding with such activities.

F. The licensee shall notify the Commission, as soon as possible but not later than one hour, of any accident at this facility which could result in an unplanned release of quantities of fission products in excess of allowable limits for normal operation established by the Commission.

G. The licensee shall have and maintain financial protection of such type and in such amounts as the Commission shall require in accordance with Section 170 of the Atomic Energy Act of 1954, as amended, to cover public liability claims.

H. This license is effective as of the date of issuance and shall expire at midnight on April 17, 2022; provided however that should the Commission, in conjunction with its consideration of the petition for Rulemaking in Docket No. PRM-50-30 determine that the termination date for operating licenses should appropriately run from the date of the issuance of a licensee's construction permit, the expiration date of this license will be September 10, 2013, effective upon notice to the Licensee of the Commission's action in this regard.

FOR THE NUCLEAR REGULATORY COMMISSION

Harold R. Denton

Harold R. Denton, Director
Office of Nuclear Reactor Regulation

Attachment:
1. Attachment 1
2. Appendix A - Technical
   Specifications (NUREG-0861)
3. Appendix B - Environmental
   Protection Plan

Date of Issuance: April 17, 1982

From November 1979 to February 1980 I was out on sick leave. When I returned to the site in February 1980 I worked as a core driller for Foley Electrical Co. until July 1980. During this time period the procedures for contacting rebar were changed. We were instructed to relocate small holes when rebar was contacted, and we were only allowed to cut through the rebar if approval was given by an engineer. Written reports were also made of each hole drilled during this time period. I stopped working at the LaSalle plant as of July 31, 1980, because of an injury.

[redacted]

SUBSCRIBED AND SWORN TO
BEFORE ME THIS [illegible] DAY
OF [illegible] 1982.

[illegible]
Notary Public

AFFIDAVIT OF DALE G. BRIDENBAUGH

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

DALE G. BRIDENBAUGH, being duly sworn, deposes and says as follows:

1. I am a Professional Nuclear Engineer, technical consultant, and a founder and president of MHB Technical Associates, technical consultants on energy and environment, with offices at [illegible]23 Hamilton Avenue, Suite K, San Jose, California. I have participated as an expert witness in licensing proceedings before the U.S. Nuclear Regulatory Commission (NRC); have served as a consultant to the NRC; have testified at the request of the Advisory Committee on Reactor Safeguards; have appeared before various committees of the U.S. Congress and testified in various state licensing and regulatory proceedings.

2. I am a graduate engineer thoroughly familiar with the design, construction, and operation of nuclear generating plants, including operational errors, equipment and system failures, and other problems that could lead to adverse safety and reliability consequences. I received a B.S. in Mechanical Engineering from the South Dakota School of Mines and Technology in 1953, and have since been registered in the state of California as a Professional Nuclear Engineer. Further details

of my experience and qualifications are contained in my resume Attachment 1.

3. The purpose of this Affidavit is to identify my concerns regarding the adequacy and quality of construction of certain structures which make up an essential portion of Commonwealth Edison Company's LaSalle Nuclear Plant. I have reviewed the Affidavit of [redacted] which describes numerous cases of anchor bolt hole drilling and conduit passageway core drilling in the LaSalle Units 1 and 2 reactor buildings during the period of June, 1978 through July, 1980. If, as is reported in [redacted] Affidavit, such drilling was conducted so that reinforcing steel in concrete walls was damaged and/or completely severed without the benefit of appropriate structural analysis, this would appear to me to be a condition with potential safety significance and one that should be thoroughly investigated at LaSalle prior to plant operation.

4. I have no way of knowing whether the reported practice has in fact jeopardized safety-related structures as I do not have access to the exact locations of the holes that were drill [redacted] Affidavit, however, indicates that such drilling practices were "usual" with the associated implication that the practice was in common use by a large number of electrical crew working throughout the plant. If the practice was widespread and used by all drillers during this time period, it seems near certain that some safety-related structures (those associated

with systems or components assuring the integrity of the reactor coolant pressure boundary or those necessary to maintain the capability to shut down the reactor and maintain it in a safe shutdown condition, or those needed to prevent or mitigate the consequences of accidents which could result in potential off-site exposures) would have been affected. If so, the associated damage or degradation of safety margins of safety-related structures would appear to have violated the quality requirements imposed by the U.S. Code of Federal Regulation, 10 CFR Part 50, Appendix A, General Design Criteria for Nuclear Power Plants and Appendix B, Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants. It is also possible, if the practice was widespread to the extent that it also was used in the attachment of components and equipment to the primary containment structure, that the integrity of that structure could be affected. The LaSalle Nuclear Plant configuration includes a Mark II concrete containment structure designed to contain and mitigate the consequences of design-basis accidents that could occur during the operation of the plant. The U.S. NRC reviews the adequacy of this containment to assure its compliance with federal regulations. Standard Review Plan 3.8.1, Concrete Containment, discusses the points normally covered by the NRC in such review. The impact of the drilling operations described in [redacted] Affidavit would be relevant to the review conducted in accordance with Part

Page 3.8.1-14, which covers materials, quality control, a special construction techniques of concrete containment. The concepts expressed in this Review Plan applying to co containment would also apply to the structural integrity other concrete safety-related walls and structures.

5. I have been informed that some of the facts cont in [redacted] Affidavit have been verbally communicat to the U.S. Nuclear Regulatory Commission as called for i 10 CFR Part 21 (U.S. Code of Federal Regulations), but th investigation has yet been reported. I have also been in that the U.S. NRC has been verbally informed (by an unide employee) that the concrete roof slab making up the ceili the LaSalle off gas building is below specified thickness contains numerous holes and cracks. I have further been informed that the NRC's response to the report of this condition was that no investigation of this condition was warranted. While it is not likely that failure of the of gas building roof would result in a "calamitous" accident it does contain equipment and components handling radioac gases. The primary significance, however, of the reporte failure to investigate this condition by the NRC is the question that it raises as to the efficacy of the entire quality oversight function conducted by the NRC on the overall construction of the LaSalle Plant. This question makes more urgent the necessity to resolve the reported

/////

deficiencies that may exist in the reactor building (and other structures).

6. Prompt action to investigate these concerns is important. It is my understanding that the LaSalle Unit 1 Operating License is about to be issued which would permit the loading of fuel into the reactor and initial operations to begin. While fuel loading in itself is not likely to chang the loading conditions of the potentially affected structures so that a failure would be expected, fuel loading does represent a point in time that is of significance in the proper conduct of the investigation that may be required. When fuel loading occurs and low power operation is possible, access to certain areas of the plant, including portions of primary containment and reactor building must be controlled and/or minimized and the free movement from compartment to compartment by investigatory personnel could be restricted. Subsequent power operation of the reactor could make physical access to some portions of the facility impossible or at leas extremely limited.

7. The consequences of the degradation of the structura quality potentially represented by the severance of reinforci steel in the concrete walls is the potential failure of the structures and/or systems to perform their safety related functions under accident or seismic conditions. In my opinio it is essential that a thorough investigation be made by the appropriate authorities of the allegations raised. This wou

assure that damage to the essential structures, if it in fact exists, has been properly analyzed by appropriate technical experts and repairs or modifications are made if needed before these safety systems are called upon to prevent or mitigate the consequences of an accident.

Dale G. Bridenbaugh

March 17, 1982

Subscribed and sworn to before me this 17 day of MARCH, 1982.

NOTARY PUBLIC

My commission expires: 8/21/84

OFFICIAL SEAL

PROFESSIONAL QUALIFICATIONS OF DALE G. BRIDENBAUGH

DALE G. BRIDENBAUGH
1723 Hamilton Avenue
Suite K
San Jose, CA 95125
(408) 266-2716

EXPERIENCE:

1976 - PRESENT

President - MHB Technical Associates, San Jose, California.
Co-founder and partner of technical consulting firm. Specialist in energy consulting to governmental and other groups interested in evaluation of nuclear plant safety and licensing. Consultant in this capacity to state agencies in California, New York, Illinois, New Jersey, Pennsylvania, Oklahoma and Minnesota and to the Norwegian Nuclear Power Committee, Swedish Nuclear Inspectorate and various other organizations and environmental groups. Performed extensive safety analysis for Swedish Energy Commission and contributed to the Union of Concerned Scientist's Review of WASH-1400. Consultant to the U.S. NRC - LWR Safety Improvement Program, performed Cost Analysis of Spent Fuel Disposal for the Natural Resources Defense Council, and contributed to the Department of Energy LWR Safety Improvement Program for Sandia Laboratories. Served as expert witness in NRC and state utility commission hearings.

1976 - (FEBRUARY - AUGUST)

Consultant, Project Survival, Palo Alto, California.

Volunteer work on Nuclear Safeguards Initiative campaigns in California, Oregon, Washington, Arizona, and Colorado. Numerous presentations on nuclear power and alternative energy options to civic, government, and college groups. Also resource person for public service presentations on radio and television.

1973 - 1976

Manager, Performance Evaluation and Improvement, General Electric Company - Nuclear Energy Division, San Jose, California.

Managed seventeen technical and seven clerical personnel with responsibility for establishment and management of systems to monitor and measure Boiling Water Reactor equipment and system operational performance. Integrated General Electric resources in customer plant modifications coordinated correction of causes of forced outages and of efforts to improve reliability and per

1956 - 1963

Field Engineer, General Electric Company. Installation and Service Engineering Department, Chicago, Illinois.

Supervised installation and maintenance of steam turbines of all sizes. Supervised crews of from ten to more than one hundred men, depending on the job. Worked primarily with large utilities but had significant work with steel, petroleum and other process industries. Had four years of experience at construction, startup, trouble-shooting and refueling of the first large-scale commercial nuclear power unit.

1955 - 1956

Engineering Training Program, General Electric Company, Erie, Pennsylvania, and Schenectady, New York.

Training assignments in plant facilities design and in steam turbine testing at two General Electric Factory locations.

1953 - 1955

United States Army - Ordnance School, Aberdeen, Maryland.

Instructor - Heavy Artillery Repair. Taught classroom and shop disassembly of artillery pieces.

1953

Engineering Training Program, General Electric Company, Evendale, Ohio.

Training assignment with Aircraft Gas Turbine Department.

EDUCATION & AFFILIATIONS:

BSME - 1953, South Dakota School of Mines and Technology, Rapid City, South Dakota, Upper ½ of class.

Professional Nuclear Engineer - California. Certificate No. 0973.

Member - American Nuclear Society.

Various Company Training Courses during career including Professional Business Management, Kepner Tregoe Decision Making, Effective Presentation, and numerous technical seminars.

1973 - 1976 (Contd)

Responsible for development of Division Master Performance Improvement Plan as well as for numerous Staff special assignments on long-range studies. Was on special assignment for the management of two different ad hoc projects formed to resolve unique technical problems.

1972 - 1973

Manager, Product Service, General Electric Company - Nuclear Energy Division, San Jose, California.

Managed group of twenty-one technical and four clerical personnel. Prime responsibility was to direct interface and liaison personnel involved in corrective actions required under contract warranties. Also in charge of refueling and service planning, performance analysis, and service communication functions supporting all completed commercial nuclear power reactors supplied by General Electric, both domestic and overseas (Spain, Germany, Italy, Japan, India, and Switzerland).

1968 - 1972

Manager, Product Service, General Electric Company - Nuclear Energy Division, San Jose, California.

Managed sixteen technical and six clerical personnel with the responsibility for all customer contact, planning and execution of work required after the customer acceptance of department-supplied plants and/or equipment. This included quotation, sale and delivery of spare and renewal parts. Sales volume of parts increased from $1,000,000 in 1968 to over $3,000,000 in 1972.

1966 - 1968

Manager, Complaint and Warranty Service, General Electric Company - Nuclear Energy Division, San Jose, California.

Managed group of six persons with the responsibility for customer contacts, planning and execution of work required after customer acceptance of department-supplied plants and/or equipment--both domestic and overseas.

1963 - 1966

Field Engineering Supervisor, General Electric Company, Installation and Service Engineering Department, Los Angeles, California.

Supervised approximately eight field representatives with responsibility for General Electric steam and gas turbine installation and maintenance work in Southern California, Arizona, and Southern Nevada. During this period was responsible for the installation of eight different central station steam turbine generator units, plus much maintenance activity. Work included customer contact, prepa-

HONORS & AWARDS:

Sigma Tau - Honorary Engineering Fraternity.

General Managers Award, General Electric Company.

PERSONAL DATA:

Born November 20, 1931, Miller, South Dakota.
Married, three children
6'2", 190 lbs., health - excellent
Honorable discharge from United States Army
Hobbies: Skiing, hiking, work with Cub and Boy Scout Groups.

PUBLICATIONS & TESTIMONY:

1. Operating and Maintenance Experience, presented at Twelfth Annual Seminar for Electric Utility Executives, Pebble Beach, California, October 1972, published in General Electric NEDC-10697, December 1972.

2. Maintenance and In-Service Inspection, presented at IAEA Symposium on Experience From Operating and Fueling of Nuclear Power Plants, Bridenbaugh, Lloyd & Turner. Vienna, Austria, October, 1973.

3. Operating and Maintenance Experience, presented at Thirteenth Annual Seminar for Electric Utility Executives, Pebble Beach, California, November, 1973, published in General Electric NEDO-20222, January. 1974.

4. Improving Plant Availability, presented at Thirteenth Annual Seminar for Electric Utility Executives, Pebble Beach, California, November 1973, published in General Electric NEDO-20222, January, 1974.

5. Application of Plant Outage Experience to Improve Plant Performance, Bridenbaugh and Burdsall, American Power Conference, Chicago, Illinois, April 14, 1974.

6. Nuclear Valve Testing Cuts Cost, Time, Electrical World, October, 15, 1974.

7. The Risks of Nuclear Power Reactors: A Review of the NRC Reactor Safety Study WASH-1400, Kendall, Hubbard, Minor & Bridenbaugh, et al, for the Union of Concerned Scientists, August, 1977.

8. Swedish Reactor Safe Study, Barsebäck Risk Assessment, MHB Technical Associates, January, 1978. (Published by the Swedish Department of Industry as Document DsI 1978:1)

9. Testimony of D.G. Bridenbaugh, R.B. Hubbard, G.C. Minor to the California State Assembly Committee on Resources, Land Use, and Energy, March 8, 1976.

10. Testimony of D.G. Bridenbaugh, R.B. Hubbard, and G.C. Minor before the United States Congress, Joint Committee on Atomic Energy, February 18, 1976, Washington, DC (Published by the Union of Concerned Scientists, Cambridge, Massachusetts.)

11. Testimony by D.G. Bridenbaugh before the California Energy Commission, entitled, Initiation of Catastrophic Accidents at Diablo Canyon, Hearings on Emergency Planning, Avila Beach, California, November 4, 1976.

12. Testimony by D.G. Bridenbaugh before the U.S. Nuclear Regulatory Commission, subject: Diablo Canyon Nuclear Plant Performance, Atomic Safety and Licensing Board Hearings, December, 1976.

13. Testimony by D.G. Bridenbaugh before the California Energy Commission, subject: Interim Spent Fuel Storage Considerations, March 10, 1977.

14. Testimony by D.G. Bridenbaugh before the New York State Public Service Commission Siting Board Hearings concerning the Jamesport Nuclear Power Station, subject: Effect of Technical and Safety Deficiencies on Nuclear Plant Cost and Reliability, April, 1977.

15. Testimony by D.G. Bridenbaugh before the California State Energy Commission, subject: Decommissioning of Pressurized Water Reactors, Sundesert Nuclear Plant Hearings, June 9, 1977.

16. Testimony by D.G. Bridenbaugh before the California State Energy Commission, subject: Economic Relationships of Decommissioning, Sundesert Nuclear Plant, for the Natural Resources Defense Council, July 15, 1977.

17. Testimony by D.G. Bridenbaugh before the Vermont State Board of Health, subject: Operation of Vermont Yankee Nuclear Plant and Its Impact on Public Health and Safety, October 6, 1977.

18. Testimony by D.G. Bridenbaugh before the U.S. Nuclear Regulatory Commission, Atomic Safety and Licensing Board, subject: Deficiencies in Safety Evaluation of Non-Seismic Issues, Lack of a Definitive Finding of Safety, Diablo Canyon Nuclear Units October 18, 1977, Avila Beach, California.

19. Testimony by D.C. Bridenbaugh before the Norwegian Commission on Nuclear Power, subject: Reactor Safety/Risk, October 2[illegible], 1977.

20. Testimony by D.G. Bridenbaugh before the Louisiana State Legislature Committee on Natural Resources, subject: Nuclear Power Plant Deficiencies Impacting on Safety & Reliability, Baton Rouge, Louisiana, February 13, 1978.

21. Spent Fuel Disposal Costs, report prepared by D.G. Bridenbaugh for the Natural Resources Defense Council (NRDC), August 31, 1978.

22. Testimony by D.G. Bridenbaugh, G.C. Minor, and R.B. Hubbard before the Atomic Safety and Licensing Board, in the matter of the Black Fox Nuclear Power Station Construction Permit Hearings, September 25, 1978, Tulsa, Oklahoma.

23. Testimony of D.G. Bridenbaugh and R.B. Hubbard before the Louisiana Public Service Commission, Nuclear Plant and Power Generation Costs, November 19, 1978, Baton Rouge, Louisiana.

24. Testimony by D.G. Bridenbaugh before the City Council and Electric Utility Commission of Austin, Texas, Design, Construction, and Operating Experience of Nuclear Generating Facilities, December 5, 1978, Austin, Texas.

25. Testimony by D.G. Bridenbaugh for the Commonwealth of Massachusetts, Department of Public Utilities, Impact of Unresolved Safety Issues, Generic Deficiencies, and Three Mile Island-Initiated Modifications on Power Generation Cost at the Proposed Pilgrim-2 Nuclear Plant, June 8, 1979.

26. Improving the Safety of LWR Power Plants, MHB Technical Associates, prepared for U.S. Dept. of Energy, Sandia Laboratories, September 28, 1979.

27. BWR Pipe and Nozzle Cracks, MHB Technical Associates, for the Swedish Nuclear Power Inspectorate (SKI), October, 1979.

28. Testimony of D.G. Bridenbaugh and G.C. Minor before the Atomic Safety and Licensing Board, in the matter of Sacramento Municipal Utility District, Rancho Seco Nuclear Generating Station following TMI-2 accident, subject: Operator Training and Human Factors Engineering, for the California Energy Commission, February 11, 1980.

29. Italian Reactor Safety Study: Caorso Risk Assessment, MHB Technical Associates, for Friends of the Earth, Italy, March, 1980.

30. Decontamination of Krypton-85 from Three Mile Island Nuclear Plant, H. Kendall, R. Pollard, & D.G. Bridenbaugh, et al, The Union of Concerned Scientists, delivered to the Governor of Pennsylvania, May 15, 1980.

31. Decontamination of Krypton-85 from Three Mile Island Nuclear Plant, H. Kendall, R. Pollard, & D.G. Bridenbaugh, et al, The Union of Concerned Scientists, delivered to the Governor of Pennsylvania, May 15, 1980.

32. Testimony by D.G. Bridenbaugh before the New Jersey Board of Public Utilities, on behalf of New Jersey Public Advocate's Office, Division of Rate Counsel, Analysis of 1979 Salem-1 Refueling Outage, August, 1980.

33. Minnesota Nuclear Plants Gaseous Emissions Study, MHB Technical Associates, for Minnesota Pollution Control Agency, September, 1980.

34. Position Statement, Proposed Rulemaking on the Storage and Disposal of Nuclear Waste, Joint Cross-Statement of Position of the New England Coalition on Nuclear Pollution and the Natural Resources Defense Council, September, 1980.

35. Testimony by D.G. Bridenbaugh and Gregory C. Minor, before the New York State Public Service Commission, In the Matter of Long Island Lighting Company Temporary Rate Case, prepared for the Shoreham Opponents Coalition, September 22, 1980, Shoreham Nuclear Plant Construction Schedule.

36. Supplemental Testimony by D.G. Bridenbaugh before the New Jersey Board of Public Utilities, on behalf of New Jersey Public Advocate's Office, Division of Rate Counsel, Analysis of 1979 Salem-1 Refueling Outage, December, 1980.

37. Testimony by D.G. Bridenbaugh and Gregory C. Minor, before the New Jersey Board of Public Utilities, on behalf of New Jersey Department of the Public Advocate, Division of Rate Counsel, Oyster Creek 1980 Refueling Outage Investigation, February, 1981.

38. Economic Assessment: Ownership Interest in Palo Verde Nuclear Station, MHB Technical Associates, for The City of Riverside, September 11, 1981.

39. Testimony of D.G. Bridenbaugh before the Public Utilities Commission of Ohio, in the matter of the Regulation of the Electric Fuel Component Contained Within the Rate Schedules of the Toledo Edison Company and Related Matters, subject: Davis-Besse Nuclear Power Station 1980-81 Outage Review, October, 1981.

40. Supplemental Testimony of D.G. Bridenbaugh before the Public Utilities Commission of Ohio, in the matter of the Regulation of the Electric Fuel Component Contained within the Rate Schedules of the Toledo Edison Company and Related Matters, subject: Davis-Besse Nuclear Power Station 1980-81 Outage Rev[illegible] November, 1981.

41. Systems Interaction and Single Failure Criterion, Phase 2 Report, MHB Technical Associates for the Swedish Nuclear Power Inspectorate (SKI), January 7, 1982.

42. Testimony of D.C. Bridenbaugh and Gregory C. Minor on behalf of Governor Edmund G. Brown Jr., before the Atomic Safety and Licensing Board, regarding Contention 10, Pressurizer Heaters, January 11, 1982.

43. Testimony of D.C. Bridenbaugh and Gregory C. Minor on behalf of Governor Edmund G. Brown, Jr. before the Atomic Safety and Licensing Board, regarding Contention 12, Block and Pilot Operated Relief Valves, January 11, 1982.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION III
799 ROOSEVELT ROAD
GLEN ELLYN, ILLINOIS 60137

Fortune
RF
Bitz

March 30, 1982

MEMORANDUM FOR: R. C. DeYoung, Director, Office of Inspection and Enforcement

FROM: James G. Keppler, Regional Administrator, Region III

SUBJECT: LA SALLE COUNTY NUCLEAR STATION - PETITION FROM ILLINOIS ATTORNEY GENERAL

As you know, on March 24, 1982, the Illinois Attorney General petitioned the NRC to suspend licensing proceedings at La Salle pending investigation of recent allegations and to institute a Show Cause Hearing with Illinois as a party to the Hearing. The allegations deal with the overall adequacy of safety related structures as a result of widespread rebar cutting and specific structural deficiencies in the roof of the off-gas building.

A conference call was held on March 29 involving Messrs. Denton, Case, Stello, DeYoung and Keppler to discuss the handling of these investigations. We agreed that, because the petition expresses concern that the off-gas building deficiencies had been verbally communicated earlier to NRC and that the NRC had concluded an investigation of these alleged deficiencies was not warranted, it would be prudent to have an independent review of this allegation by IE (since IE was not involved in the consideration not to investigate). This review should address both the technical adequacy of the off-gas building concerns as well as the NRC's handling of the earlier notification in this regard. With respect to the concerns associated with cutting through rebar this matter will be reviewed by Region III with technical assistance from NRR.

I realize your staff is already depleted as a result of other investigation assistance you are giving us, and your willingness to assist in this effort is genuinely appreciated.

James G. Keppler
James G. Keppler
Regional Administrator

cc: V. Stello, DEDROGR
H. R. Denton, NRR

8205060155

PDR/LPDR

A22/3

ATTACHMENT (2)

A22/3

((Warnick
(File Chas Copy)

January 28, 1982.

MEMORANDUM FOR: Region III Files - LaSalle

FROM: Robert F. Warnick, Director, Enforcement and Investigation Staff

SUBJECT: TELEPHONE CALL FROM DOUG LONGENIE OF CHANEL 5 TV ALLEGATIONS AT LA SALLE

Doug Longenie called on January 26, 1982 at about 2 p.m. regarding information he had been given while pursuing other news stories. Doug indicated the source of his information was [redacted] Longenie indicated he would contact [redacted] on January 27, 1982, and pave the way for us to contact [redacted] to get specific details of the following allegations:

1. [redacted] claimed that when they were putting in conduit, the grounding wasn't adequate. They did not do a good job of grinding off the zinc.

2. [redacted] knows where two radiation monitors were supposed to be installed in the off gas building but were not actually installed.

3. All of the required heating pads in the off gas building were not installed.

4. The ceiling (in the off gas building, I think) was supposed to be 12 inches thick. When they were drilling 8 inch anchor bolts, they penetrated the ceiling and could see sky.

5. The fire alarm system in reactor building No. 1 does not meet specifications. A CAR was issued but it has been written off without the work being done. This problem was also mentioned by a second individual but the second person will not talk to us about it.

A22/4

~~8206090242~~
PDR

RIII	RIII					
	Warnick					

6. Dust seals were only pently installed.

I told Longenie we would get in touch with [redacted] and follow up on these allegations.

Robert F. Warnick, Director
Enforcement and Investigation Staff

cc: A. B. Davis
C. E. Noreiius
L. R. Spessard

OFFICE						
SURNAME						

ATTACHMENT (3)

*A22/4

Warnick (file - ...)

~~DO NOT DISCLOSE~~
~~Contains identity of~~
~~confidential source~~

FEB 10 1982

MEMORANDUM FOR: Region III Files

FROM: James E. Foster, Investigator

SUBJECT: TELEPHONE CONTACT RE LA SALLE ALLEGATIONS (Ref. Warnick Memo of 1/28/82)

I contacted [redacted] on February 4, 1982, at approximately 7:30 p.m. [redacted] indicated that Dough Longenie, the Channel 5 newsman who provided [redacted] name to Warnick, had not advised [redacted] that [redacted] would be contacted by Region III. Nevertheless [redacted] was very willing to discuss [redacted] concerns with me, at length. [redacted] requested confidentiality.

[redacted]

I discussed each point enumerated in the Warnick memo, and developed the following information:

1. Conduit grounding was not properly done for most LaSalle construction. During the late phase of construction, [redacted] had read the grounding specification, questioned a QC inspector regarding specification requirements, pointed out deficiencies, and grounding was properly performed from that point on. Prior to that time, crews had not cleaned and copper coated conduit threads nor adequately ground off the zinc conduit coatings where grounding straps were attached. [redacted] estimated that some 80% of the installed conduit was not properly grounded (per specification, developed from a NEMA requirement).

2. In the off-gas building, there is a location where radiation sensors for Unit 1 and Unit 2 are in close proximity. This was described as being at the 710 foot elevation, East of AB wall, between 14 and 13 line in the filter building (part of the off-gas building). The sensor that is not installed is for Unit 2. [redacted] feels that the Unit 2 sensor should be installed now, as the location will be radioactive and the installation difficult after Unit 1 is in operation.

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SURNAME	Foster/gg	Warnick RW					
DATE	2/19/82						

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3. [redacted] stated that approximately one month ago, five heating pads were removed from the re-heat cylinders located in the off-gas building. These re-heat cylinders are reached via an entrance on the 710' elevation, and then by going down two elevations. The heating pads were removed to allow some work being performed by fitters, and [redacted] believes that the pads were not replaced when the area was "closed up".

4. Holes drilled for expansion anchors in the ceiling of the off-gas building (725 foot elevation) penetrated the concrete and asphalt roof covering. There was water accumulation on the roof, and water came in via the anchor bolts. There are cracks in the concrete between holes because the holes were drilled too close to each other. This was brought to the attention of [redacted] from Sargent & Lundy, and some patching was performed.

5. The fire detector modules have been wired without regard to separation criteria. The crews were wiring the detectors from any hanger indiscriminately.

6. Dust protection was not installed on conduits and conduit boxes as specified for dust protection. Some have seal gaskets but no o-rings. [redacted] observed this during installation of the security system wiring, and the required seals were installed after [redacted] brought this to the attention of Bill Bags.

In addition to the above information (which expands on that provided by Doug Longenie), [redacted] indicated that [redacted] was in charge of an area which had nonconformances written on some of the equipment, but the work was not corrected as indicated in the nonconformance report close-out. [redacted] had the nonconformances approved as completed when the regular inspector was absent.

[redacted] also indicated that some core-drill sheets were found to have information which had been whited-out. (This may be related to the TV-5 story "drilling for dollars" which aired on the 10 o'clock news, February 4, 1982. [redacted] stated that Longenie had advised [redacted] that the story would air that evening.)

James E. Foster

James E. Foster
Investigator

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ATTACHMENT (4)

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1. Warnick

February 26, 1982

MEMORANDUM FOR: Region III Files - LaSalle

THRU: Robert F. Warnick, Director, Enforcement and Investigation Staff

FROM: G. A. Phillip, Investigator

SUBJECT: TELEPHONE CONTACT FROM MRS. JUDITH GOODIE, ATTORNEY, ILLINOIS ATTORNEY GENERAL'S OFFICE

On February 22, 1982 I was informed that Mrs. Judith Goodie, an attorney in the Illinois Attorney General's office (312-793-2491), had called concerning allegations made by [redacted] via TV Channel 5 reporter Doug Longenie. Information regarding these allegations is contained in two memoranda to Region III Files, one prepared by R. F. Warnick dated January 28, 1982, and the other prepared by J. E. Foster dated February 10, 1982.

Before speaking with Mrs. Goodie I attempted to obtain additional information regarding the allegations and any action we had taken or planned to take. The following information was obtained, primarily through discussions with Roger Walker, regarding the items listed in the above memoranda.

1. Conduit grounding is an industrial safety not a nuclear safety concern and therefore need not be pursued by NRC.

2. Since it is not known whether Unit 2 will be built, the NRC cannot force the licensee to take action to install a Unit 2 sensor.

3. The absence of the heating pads would become apparent during pre-op testing. This is a radiation safety concern. In all likelihood the licensee has a means of tracking this matter to assure that the pads are replaced. We could with little effort confirm this.

4. This matter is not of concern to NRC since this structure is not considered safety related, i.e. subject to seismic considerations.

5. No additional information obtained.

6. No additional information obtained.

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SURNAME	Phillip/gg	Warnick				
DATE						

During a telephone conversation with Goodie on February 23, 1982, I discussed the above with her and indicated we were evaluating the allegations to decide what action we would take. She indicated she would call Foster or me on March 1 or 2 to find out what we planned to do.

Goodie indicated she had talked with [redacted] and asked whether [redacted] had made an allegation during our contacts with [redacted] regarding the drilling into rebar which [redacted] said had occurred during the early stages of construction. I indicated that I was unaware of that allegation but that it would be of interest to us.

Following my conversation with Mrs. Goodie I was advised by Walker that the LaSalle Resident Inspector had determined that the heating pad removal was documented on Pre-operational Deficiency No. 402 and they were required to be replaced prior to fuel load.

Regarding Items 5 and 6, Ron Gardner advised that we do not inspect security equipment wiring. I believe it is NRC's position that security equipment must function as required by the licensee's security plan commitments and if they fail compensatory measures must be taken until they are repaired and operational.

On the basis of the above it appears that there is no need to further pursue the matters [redacted] has brought to our attention. This should be conveyed to [redacted] and at the same time we should attempt to determine whether [redacted] has information regarding the drilling into rebar that warrants further action.

Gerald A. Phillip
Investigator

cc: R. Walker
J. Creed
R. Gardner
J. Foster

ATTACHMENT (5)

*A22/6

EIS File No. 8205

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3/13/82

MEMORANDUM FOR: Region III Files

THRU: Robert F. Warnick, Director, Enforcement and Investigation Staff

FROM: James E. Foster, Investigator

SUBJECT: ALLEGATION RE REBAR CUTTING AT LA SALLE, DOCKET NO.50-373 (REF. PHILLIP MEMO OF 2/26/82)

On March 6 and 8, 1982, I was contacted by Ms. Judith Goodie, of the Illinois Attorney General's Office. She indicated that she had been in contact with [redacted] regarding [redacted] allegations concerning work [redacted] at the LaSalle site. Ms. Goodie indicated [redacted] had told her that [redacted] had often cut reinforcement bars (rebar) when drilling cores or holes at LaSalle. She also indicated she felt that [redacted] concerns regarding equipment in the off-gas building should not be dismissed, as some equipment in the building was intended to reduce or mitigate radioactive releases during an accident.

I advised Ms. Goodie that [redacted] had not made any allegations to me regarding cutting of rebar during our previous conversations, and I would try to recontact [redacted] I also advised that the off-gas building was a non-safety, non-siesmic structure, and as such should not contain safety-related equipment (NRC definition of safety-related). She indicated she had talked to "nuclear experts" who had advised her differently.

I recontacted [redacted] at approximately 9:18 p.m. on March 8, 1982. [redacted] stated that, up until approximately a year and a half ago (September 1980?) when [redacted] hit a rebar during core drilling, a special crew was called. This crew operated a special, water cooled, diamond drill rig, which would cut the rebar. This was referred to as a "wet hole" due to the water cooling, and the utility was billed for the extra work entailed (crew, diamond drill use, laborers to clean up the water).

[redacted] indicated [redacted] worked in the reactor building and in the off-gas building, but the problem was "generic" to LaSalle, as other crews also followed the same practice in other areas. [redacted] stated that a notice ("work notice") was finally put out by Quality Control stopping the uncontrolled cutting.

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RIII	RIII
Foster/gg	Warnick
3/12/82	3/13/82

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I inquired if any present workers would be able to provide Region III with additional information, and [redacted] indicated that a [redacted] may still be onsite, and may be able to provide additional information and locations of cut rebar.

It appears that the allegation can be checked by a review of billing records for core drilling. Those with additional charges for the diamond drill crew should provide locations where reinforcement bars were cut by the drilling. It should also be relatively easy to locate a work notice informing the crews that uncontrolled rebar cutting was to stop.

As the LaSalle plant will be ready for operating license issuance in the near future, I recommend that this issue receive priority attention.

James E. Foster
Investigator

cc: L. Spessard
R. Walker
R. Gardner
C. E. Norelius

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ATTACHMENT (6)

* A22/7

UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION III
799 ROOSEVELT ROAD
GLEN ELLYN, ILLINOIS 60137

MAR 25 1982

3/31 - Spessard
Heup
Walker
Warnick
file (Julie)

MEMORANDUM FOR: Charles E. Norelius, Director, Division of Engineering and Technical Programs

FROM: Robert F. Warnick, Director, Enforcement and Investigation Staff

SUBJECT: ALLEGATION RE REBAR CUTTING AT LA SALLE-DOCKET NO. 50-373; 50-374

The four attached memos document concerns expressed by an alleger and Ms. Judith Goodie of the Illinois Attorney General's office regarding core drilling through rebar.

Because of the high priority of LaSalle and the unavailability of investigators, this matter is being transferred to your Division as we discussed on March 22, 1982.

EIS would appreciate receiving a copy of the documentation of your findings and closeout.

Robert F. Warnick, Director
Enforcement and Investigation Staff

Attachments: As stated

cc w/attachments:
R. L. Spessard
R. Walker
J. Foster

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PDR

A22/8

Attention
DRF I. FILE COPY

ASSISTANCE REQUEST FORM

To: Charles E. Norelius - ____________ (Assigned) Date: 3/25/82

From: Robert F. Warnick - ____________ (Requestor) Requested Completion Date: ____________

Facility: LaSalle Docket No.: 50-373 50-374 Category: ____________

Request Description: Followup on allegations & A copy of the documentation of your findings and closeout

Reference: See attached Memos From Warnick, Foster(?) & Phillips attached (R2 - R5)

Detail: See attached Cons. drilling three Rebar

/s/ RLW
Section Chief

CLOSEOUT

Reference:

Detail:

(Date) /S/ ____________ Section Chief

A22/8

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Attachment 1

ATTACHMENT (7)

A22/8

UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION III
799 ROOSEVELT ROAD
GLEN ELLYN, ILLINOIS 60137

EIS File No. 82-05

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MAR 31 1982

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MEMORANDUM FOR: Region III Files

FROM: James E. Foster, INvestigator

THROUGH: R. W. Warnick, Director, Enforcement and Investigation Staff

SUBJECT: CONTACT WITH JUDITH GOODIE

I contacted Ms. Judith Goodie, of the Illinois Attorney General's Office, at approximately 9:10 a.m., on March 26, 1982.

I advised Ms. Goodie that Region III had not been aware of allegations by [redacted] regarding LaSalle, and inquired why the Illinois Attorney General's petition did not mention allegations from [redacted]

Ms. Goodie stated that she had "assumed" that Region III had gotten [redacted] name from NBC (as she had) and had contacted [redacted] She indicated that [redacted] had declined to provide her office with an affidavit for fear of [redacted] name being known, and so was not included in the submitted petition.

I indicated that the information provided by [redacted] was much more detailed than that provided by [redacted] and would have assisted Region III in its review.

Ms. Goodie stated that she had not meant to withhold any information, and that "it should have been obvious that we were working on something".

James E. Foster

James E. Foster
Investigator

cc: R. L. Spessard
R. Gardner
W. Walker
C. E. Norelius

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ATTACHMENT (8)

*A22/9

Roger Walker

March 29, 1982

OFF-GAS BUILDING ROOF REPORT

PURPOSE

The purpose of the report is to state information regarding the second allegation (Page 6, Request to Institute a Show Cause Proceeding and for other Relief - Tyrone Fahner, Attorney General of the State of Illinois) on the Off-Gas Building roof. The evidence shows that the allegation is false.

BACKGROUND

The concrete enclosure above grade as part of the Off-Gas Building is a non-safety related structure which houses Off-Gas Building HVAC Air Handling Units, HVAC Water Cooled Condensing Units, HVAC Exhaust Filter Units, HVAC Control Panels and associated motor control centers and switchgear. The specification concrete compressive strength is 4000 psi at 90 days. While detailed quality assurance requirements were not required due to the building being non-safety related, they were applied as part of the overall Commonwealth Edison / Walsh Construction Company quality effort.

FINDINGS

The Off-Gas Building enclosure concrete (walls and roof) was poured on November 7, 1975. Walsh Construction Company(WCC) Q.C. Form QCP-9A(Pour Checkout Card) was signed by the appropriate construction and Q.C. personnel and countersigned by a Commonwealth Edison Company Field Engineer. Additionally, WCC Q.C. Forms QCP-6A(Reinforcing Steel Placement Audit) and QCP-93(Concrete Placement Control Audit Form) were utilized and signed by WCC Q.C. personnel. Concrete testing during the pour by A & H Engineering Corporation showed the concrete was within specification requirements for slump, air content and placing temperature. The concrete met compressive strength requirements, the lowest cylinder break was 4670 psi at 90 days.

On September 25, 1979, Commonwealth Edison Company Quality Assurance pointed out some surface cracking in the bottom of the Off-Gas Building roof. The area had a high density of concrete expansion anchors. An inspection performed by WCC Q. A. Supervisor, WCC General Superintendant and CECO Structural Engineer found the cracking to be surface in nature and no further action required.

A temporary construction power center transformer and switchgear were set on the roof in 1976. The unit weighed approximately 6700 pounds. The unit was set over a concrete beam in the longitudual direction and one end rested on the east concrete wall. A check was made to insure the roof would take the unit loading prior to installation. The unit was removed in late 1981 as it was no longer required.

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The slab thickness has been checked on two different occasions. On March 10, 1982 a single point check showed the slab as 1' - 2½" thick including roofing material. Roofing material is approximately 1-3/4" - 2" thick. Additional slab thickness checks were made on March 29, 1982. Fifteen (15) points checked showed the slab plus roofing materail varied from 1' - 1-5/8" to 1' - 3-3/4". A check made by effectively eliminating the roofing material showed the slab thickness varied from 11-1/4" to 1' 1-1/4".

A visual survey of the roof underside was made by KCC Q.A. and CECO on March 27, 1982. The survey showed no abnormal concrete cracking. The area under the former electrical equipment showed no abnormal concrete cracking.

SUMMARY

The Off-Gas Building roof concrete is 12 inches thick per specifications. There is no abnormal concrete cracking due to concrete expansion anchors and/or the former electrical equipment. The roof will serve its' intended function.

Prepared by:

Daniel L. Shamblin

Daniel Shamblin

ATTACHMENT (9)

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