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

ComEd

February 15, 2000

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

LaSalle County Station, Units 1 and 2
Facility Operating License Nos. NPF-11 and NPF-18
NRC Docket Nos. 50-373 and 50-374

Subject: Response to Request for Additional Information
License Amendment Request for Power Uprate Operation

References: (1) Letter from R. M. Krich, Commonwealth Edison (ComEd) Company, to U.S. NRC, "Request for License Amendment for Power Uprate Operation," dated July 14, 1999.

(2) Letter from D. M. Skay, U.S. NRC, to Commonwealth Edison (ComEd) Company, "Request for Additional Information – LaSalle County Station, Units 1 and 2 (TAC Nos. MA6070 and MA6071) (the letter contains 14 questions), dated December 27, 1999.

In the Reference 1 letter, pursuant to 10 Code of Federal Regulations (CFR) 50.90, we proposed to operate both LaSalle County Station Units at an uprate power level of 3489 MWT. In the Reference 2 letter, the NRC requested additional information concerning the proposed amendment to support their review. The attachment to this letter provides our response to the request for additional information.

The no significant hazards consideration, submitted in Reference 1, remains valid for the information attached.

A001

Should you have any questions concerning this letter, please contact Mr. Frank A. Spangenberg, III, Regulatory Assurance Manager, at (815) 357-6761, extension 2383.

Respectfully,

for

Jeffrey A. Benjamin
Site Vice President
LaSalle County Station

Attachment

cc: Regional Administrator – NRC Region III
NRC Senior Resident Inspector – LaSalle County Station

Attachment
Response to Request for Additional Information

STATE OF ILLINOIS	)	
IN THE MATTER OF	)	
COMMONWEALTH EDISON COMPANY	)	
LASALLE COUNTY STATION - UNIT 1 & UNIT 2	)	Docket Nos. 50-373 50-374

Subject: Response to Request for Additional Information License Amendment Request for Power Uprate Operation

AFFIDAVIT

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

for

Jeffrey A. Benjamin
Site Vice President
LaSalle County Station

OFFICIAL SEAL
GEORGENE R. VAN DUYNE
NOTARY PUBLIC, STATE OF ILLINOIS
MY COMMISSION EXPIRES 9-28-2003

Subscribed and sworn to before me, a Notary Public in and for the State above named, this 15th day of February, 2000.
My Commission expires on 9-28-2003.

Georgene R Van Duyne
Notary Public

Attachment
Response to Request for Additional Information

Question 1:

Section 3.3.2 of the referenced topical report states that the load combinations for the current licensing basis of the reactor vessel and internals are the reactor internal pressure difference, main steam line and recirculation line break loss-of-coolant accident (LOCA) loads, seismic and fuel lift loads. Provide an explanation why the asymmetric pressurization loads and the thrust jet loads that are increased for the power uprate were not included in the load combinations for evaluation of the reactor vessel and internal components.

Reference:

Letter, Commonwealth Edison Company to U.S. NRC, "LaSalle County Station, Units 1 and 2 - Docket Nos. 50-373 and 50-374, Request for License Amendment for Power Uprate Operation," dated July 14, 1999 - Attachment E: General Electric Nuclear Energy, Licensing Topical Report NEDC-32701P, Revision 2, "Power Uprate Safety Analysis Report for LaSalle County Station, Units 1 and 2," dated July 1999 (Proprietary).

Response 1:

The loads that have increased were considered in the evaluations performed.

The asymmetric pressurization (AP) loads were calculated before power uprate using the design basis mass and energy releases, which are still valid and bounding for power uprate. It is therefore concluded that AP loads do not change with power uprate.

The Reactor Recirculation (RR) line break thrust loads on the Reactor Pressure Vessel (RPV) are calculated based on the limiting temperature and pressure of the RR suction piping as it exits the RPV. Under power uprate conditions the changes in the RR system temperature and pressure result in a small decrease of less than 1% and thus have a negligible impact on the magnitude of the RR jet thrust loads on the RPV.

Both the AP and line break thrust loads were considered in appropriate load combinations, and the governing load combinations were used for the evaluation of the reactor vessel and internal components.

Question 2:

In Section 3.3.2.2 of the referenced topical report, an assessment of flow-induced vibration of the reactor internal components due to power uprate is performed to address the increase in steam product in the core, the increase in the core pressure drop, and the increase in the recirculation pump speed. In that assessment, the vibration levels were estimated by extrapolating the recorded vibration data at LaSalle, Unit 1, and by using the General Electric (GE) Nuclear Energy operating experience.

a. Provide a sample evaluation and the basis for using the operating experience data.

b. Section 3.3.2.2 states that "the calculations for power uprate conditions confirm that vibrations of all safety related reactor internal components are within the GE acceptance criteria...." Please describe the components evaluated and the GE acceptance criteria.

c. Provide a sample of the highest-calculated internal component values and the corresponding GE allowable values.

Response 2:

Background:

LaSalle is a BWR/5 reactor with a 251-inch diameter vessel. Tokai 2 is the NRC designated prototype for BWR/5 251 plants. The reactor internals of LaSalle Unit 1 and LaSalle Unit 2 are identical except for the jet pump components. The LaSalle Unit 1 jet pumps have a diffuser adapter, whereas the prototype plant and LaSalle Unit 2 do not have a diffuser adapter. The reactor internals at the prototype plant were extensively instrumented during the startup testing of the plant for purposes of vibration monitoring to confirm the structural integrity of major components in the reactor with respect to flow induced vibration. Due to the diffuser adapter difference in the jet pump, the LaSalle Unit 1 jet pumps also were instrumented with strain gages and accelerometers, and extensive data was collected at various core flows along two rod lines.

At both plants extensive vibration measurements were made over a period of 2 years covering a wide range of operational conditions from pre-operational (without fuel), pre-critical (with fuel but not critical) and power operational tests. The power operational tests were conducted at two rod line conditions at various core flows. The sensor signals were recorded on-line during the test program. The vibration signals of the components were analyzed at balanced flow test conditions and at the two rod lines to determine the expected vibration response in the power uprate region. The extrapolated vibration peak amplitude response in the power uprate region was compared with the GE allowable design criteria of 10,000 psi peak stress intensity to determine the acceptability of the vibration level.

Question 2a:

Provide a sample evaluation and the basis for using the operating experience data.

Response 2a(1):

Detailed Sample Evaluation:

A sample evaluation is shown for the LaSalle Unit 1 jet pump in Table 2-1 below. The jet pumps were instrumented during the start-up testing of the plant for the purpose of vibration monitoring to confirm their structural integrity with respect to flow induced vibration. The signals of the components were analyzed at balanced flow test conditions at 75% and 100% rod lines to determine the expected vibration response in the power uprate region. The extrapolated vibration peak amplitude response in the power uprate region was compared with the GE allowable design criterion of 10,000 psi peak stress intensity. At this stress level, sustained operation is allowed without incurring any fatigue usage.

In order to apply the vibration criteria, a dynamic structural analysis is performed to relate peak stresses to measured strains or displacements at sensor locations. Mathematical models for each component are developed using finite element methods. Natural frequencies and modes of vibration are calculated. The location of the peak stress intensity is identified, including the effects of stress concentration factors. The modal strains and displacements at sensor locations are determined relative to peak stress intensity on a normalized basis, such that the highest peak stress intensity is 10,000 psi. The contribution of the various modes are absolute summed for conservatism.

At LaSalle Unit 1, there were 88 test conditions at various core flows and different rod lines. Of these, the vibrations from Increased Core Flow (IC) test condition IC-6 at 75% rod line and test condition 1 at 100% rod line were first extrapolated to 105% core flow by square extrapolation (vibration varies as the square of the core flow). Test condition IC-6 was conducted at a core flow of 105.2%, which is slightly higher than the evaluated condition of 105%. Test condition 1 was conducted at 99.7% core flow and the vibrations were extrapolated to 105% core flow by using square extrapolation. From the vibration levels at 105% core flow at these two power levels, linear extrapolation was done to the 105% power level. The detailed calculations of this evaluation are shown in Table 2-1 below, for jet pump strain gage S1.

The analysis is very conservative for the following reasons:

- The GE criterion of 10,000 psi is more conservative than the ASME criterion of 13,600 psi.
- The modes are absolute summed.
- The maximum vibration amplitude in each mode is used in the absolute sum process whereas in reality the vibration amplitude fluctuates.

For some frequencies there is a reduction in vibration with power level, but no credit is taken for this.

<u>Response 2a(2):</u>

Basis For Using Operating Experience Data:

The LaSalle units belong to the BWR/5 family with the 251- inch vessel diameter size. There are two other plants in the BWR/5-251" design which have operated at 105% uprated power conditions (WNP-2 and Nine Mile Point 2). The reactor internals of these plants are substantially identical to LaSalle Units. Since the rated core flow, steam flow, and power rating of these plants are the same as the LaSalle Units, their operating experience at uprated conditions provide additional confirmation for the conclusion that the flow induced vibrations at the LaSalle Units will remain within acceptable limits.

TABLE 2-1: SAMPLE EXTRAPOLATION FOR JET PUMP VIBRATION DATA, SENSOR S1

(1)	(2)	(3)	(4)	(5)	(6)	(7)	(8)	(9)
	Vibration at 72.7% Power			Vibration at 99.5% Power			Vibration at 105% Power	
Vibration Mode	Test Condition IC-6	Extrapolation To	% Criteria at	Test Condition # 1	Extrapolation To	% Criteria at	Extrapolation To	% Criteria at
Frequency Hz	72.7% P 105.2% CF	72.7% P 105% CF	72.7% P 105% CF	99.5% P 99.7% CF	99.5% P 105% CF	99.5% P 105% CF	105%% P 105% CF	105% P 105% CF
	microstrains p-p	microstrains p-p	(% of 10,000 psi)	microstrains p-p	microstrains p-p	(% of 10,000 psi)	microstrains p-p	(% of 10,000 psi)
18-21	41.6	41.5	21.9	36	39.9	21.1	39.9	21.1
36	15.2	15.2	10.0	12	13.3	8.8	13.3	8.8
42	10.2	10.2	5.1	12	13.3	6.7	13.9	7.0
140-175	65	64.8	45.3	65	72.1	50.4	73.6	51.4
	Absolute Sum of All Modes = 82.3% (8230 psi)			Absolute Sum of All Modes = 87% (8700 psi)			Absolute Sum of All Modes = 88.3% (8830 psi)	

Explanation of Table Contents

Column (1) lists the various measured frequencies of vibration of the jet pump.

Column (2) lists the measured magnitudes of vibration at the corresponding frequencies during 72.7% power (P) and 105.2% core flow (CF).

Column (3) lists the Column (2) values interpolated to 105% core flow using the square law.

Column (4) lists the resulting maximum stress due to each mode expressed as a percent of 10,000 psi. The stresses are absolute summed and shown in the bottom.

Column (5)* lists the measured magnitudes of vibration at the corresponding frequencies during 99.5% power and 99.7% core flow.

Column (6)* lists the Column (5) values extrapolated to 105% core flow using the square law.

Column (7)* lists the resulting maximum stress due to each mode expressed as a percent of 10,000 psi. The stresses are absolute summed and shown in the bottom.

Column (8) lists the Column (3) values (at 72.7% power) and Column (6) values (at 99.5% power) linearly extrapolated to 105% power. For frequencies in the 18-21 Hz range and at 36 Hz, the vibrations decrease with power, but no credit is taken for this and the measured value at 99.5% power is used.

Column (9) lists the resulting maximum stresses due to each mode expected at 105% power expressed as a percent of 10,000 psi. The stresses are absolute summed and shown in the bottom. Expected vibratory stress at 105% power = 8830 psi peak (less than GE acceptance criterion of 10,000 psi peak).

* Columns (5), (6) and (7) are identical in nature to columns (2), (3) and (4), except that the values correspond to 99.5% power.

Question 2b:

Section 3.3.2.2 states that "the calculations for power uprate conditions confirm that vibrations of all safety related reactor internal components are within the GE acceptance criteria...." Please describe the components evaluated and the GE acceptance criteria.

Response 2b(1):

Components Evaluated:

The critical reactor internal components affected by flow-induced vibration, which were originally identified and instrumented in the original design basis in conformance with NRC Regulatory Guide 1.20, "Comprehensive Vibration Assessment Program for Reactor Internals During Preoperational and Initial Startup Testing," (Revision 2, May 1976) were evaluated. In addition, the components, which have encountered problems in the field due to flow induced vibrations, were evaluated. The following components that were determined to require a specific assessment were evaluated:

- Shroud head and separator assembly
- Feedwater sparger
- Jet pumps
- Control rod guide tubes
- In-core guide tubes
- Liquid control/core delta p line
- Steam dryer
- Jet pump sensing lines

Response 2b(2):

GE Acceptance Criteria:

The GE criterion for stainless steel components is that the peak vibration stress intensity shall not exceed 10,000 psi. The analysis is conservative for the following reasons:

- The GE criterion of 10,000 psi peak stress intensity for stainless steel is more conservative than the ASME criterion of 13,600 psi.

- The maximum response from individual modes are absolute summed. This is conservative because the maximum modal responses generally do not occur at the same time.

Question 2c:

Provide a sample of the highest-calculated internal component values and the corresponding GE allowable values.

Response 2c:

Stress Value and Allowable Criteria:

A sample of the highest-calculated internal component value for LaSalle Unit 1 is 8830 psi peak for one jet pump occurring at the riser brace to vessel weld. The corresponding GE allowable value is 10,000 psi peak.

Question 3:

Section 3.5 for the reactor coolant pressure boundary (RCPB) piping states that the design adequacy evaluation results show that the requirements of American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (Code), Section III, Subsection NB/ND, are satisfied based on current design and licensing Code of record for the piping systems evaluated. It also states that the quantitative evaluation confirming the qualitative results will be completed prior to implementation of the uprated conditions.

a. Provide the methodology and assumptions used for evaluating the reactor coolant piping and supports for the power uprate.

b. Provide the calculated maximum stresses and fatigue usage factors, critical locations of piping systems and supports evaluated, allowable stress limits, and the Code and Code edition used in the evaluation for the power uprate.

c. Provide similar information for the balance-of-plant piping systems evaluated as listed in Section 3.11.

Question 3a:

Provide the methodology and assumptions used for evaluating the reactor coolant piping and supports for the power uprate.

Response 3a:

Existing design basis documents, such as piping stress reports, were reviewed to determine the design and analytical basis for reactor coolant and large bore balance of plant piping systems. The power uprate parameters of piping systems (pressure, temperature & flow) were compared with the existing analytical basis to determine increases in temperature, pressure, and flow due to power uprate conditions.

ASME Code, Section III, equations were reviewed to determine the equations impacted by temperature, pressure, and flow increases due to power uprate conditions.

Methodologies as described in NEDC-31897P-A, "Generic Guidelines for General Electric Boiling Water Reactor Power Uprate" (LTR1), Section 5.5.2 and Appendix K, and NEDC-31984P, "Licensing Topical Report – General Evaluations of General Electric Boiling Water Reactor Power Uprate –Supplement 2" (LTR2), Section 4.8, were used to determine the percent increases in applicable ASME Code stresses, displacements, cumulative usage factor (CUF), and pipe interface component loads (including supports) as a function of percentage increase in pressure, temperature, and flow due to power uprate conditions. The percentage increases were applied to the highest calculated stresses, displacements, and the CUF at applicable piping system node points to conservatively determine the maximum power uprate calculated stresses, displacements and usage factors. This approach is conservative because power uprate does not affect weight and dynamic loads; e.g., seismic loads are not affected by power uprate. The factors were also applied to nozzle load, support loads, penetration loads, valves, pumps, heat exchangers and anchors so that these components could be evaluated for acceptability, where required. Detailed evaluations were performed of LaSalle piping subsystems for the increases in flow, temperature, and pressure to determine the acceptability of piping support, equipment nozzle, penetration, valve, and anchor loads. The vibration displacement factor is calculated as the square of percentage increase in flow times percentage increase in pressure. No other assumptions were made to evaluate the reactor coolant piping and supports for power uprate.

Question 3b:

Provide the calculated maximum stresses and fatigue usage factors, critical locations of piping systems and supports evaluated, allowable stress limits, and the Code and Code edition used in the evaluation for the power uprate.

Response 3b:

The following LaSalle piping subsystems are impacted by 105% power uprate:

TABLE 3B-1: PIPING SUBSYSTEMS IMPACTED BY UPRATE		
Unit 1	**Unit 2**	**Description**
1FW01	2FW01	Main feedwater piping from the reactor vessel to the containment penetration (1/2FW03 after the penetration)
1FW02	2FW02	Main feedwater piping from the reactor vessel to the containment penetration (1/2FW04 after the penetration)
1FW03	2FW03	Main feedwater piping (continuing from 1/2FW01) from the containment penetration to 1/2FW05
1FW04	2FW04	Main feedwater piping (continuing from 1/2FW02) from the containment penetration to 1/2FW05
1FW05	2FW05	Main feedwater piping from 1/2FW03 and 1/2FW04 to the feedwater pumps, feedwater heaters, and condenser
1FW07	2FW07	Feedwater piping from reactor water cleanup to 1/2FW03 and 1/2FW04
1MS01	2MS01	Main steam piping from the reactor to the containment penetration (1/2MS05 after the penetration)
1MS02	2MS02	Main steam piping from the reactor to the containment penetration (1/2MS06 after the penetration)
1MS03	2MS03	Main steam piping from the reactor to the containment penetration (1/2MS07 after the penetration)
1MS04	2MS04	Main steam piping from the reactor to the containment penetration (1/2MS08 after the penetration)
1MS05	2MS08	Main steam piping (continuing from 1/2MS01) from the containment penetration to the turbine building wall penetration
1MS06	2MS07	Main steam piping (continuing from 1/2MS02) from the containment penetration to the turbine building wall penetration
1MS07	2MS06	Main steam piping (continuing from 1/2MS03) from the containment penetration to the turbine building wall penetration
1MS08	2MS05	Main steam piping (continuing from 1/2MS04) from the containment penetration to the turbine building wall penetration
1MS25	2MS25	Main steam isolation valve leak-offs to containment penetration
1RH77K (vent)	2RH77K (vent)	From reactor head vent piping to 1/2MS01 and 1/2MS02

The maximum stress ratios for each of the piping subsystems impacted by power uprate are provided below. All stresses are within the applicable ASME Code allowable stress.

TABLE 3B-2: MAXIMUM STRESS RATIOS

Subsystem	Location	Condition	Stress	Allowable	Ratio to Allowable
1FW01	66	Emergency / Faulted	34764	58410	0.60
1FW02	10A	Emergency / Faulted	36172	58410	0.62
1FW03	95	Emergency / Faulted	15595	66720	0.23
1FW04	90	Emergency / Faulted	11913	66720	0.18
1FW05	260	Thermal (Eqn. 10)	19840	22500	0.88
1FW07	125	Thermal (Eqn. 10)	12517	22500	0.56
1MS01	225T	Emergency / Faulted	25039	39825	0.629
1MS02	110T	Emergency / Faulted	30149	39825	0.76
1MS03	45T	Emergency / Faulted	28207	39825	0.71
1MS04	155T	Emergency / Faulted	20644	39825	0.52
1MS05	23	Emergency / Faulted	10184	60639	0.168
1MS06	23	Emergency / Faulted	9688	45479	0.21
1MS07	23	Emergency / Faulted	9577	45479	0.21
1MS08	23	Emergency / Faulted	9463	45479	0.21
1MS25	95	Emergency / Faulted	28820	46124	0.62
1RH77K	1227	Emergency / Faulted	33317	53100	0.63
2FW01	66	Emergency / Faulted	34764	58410	0.60
2FW02	113	Emergency / Faulted	31190	58410	0.53
2FW03	100	Emergency / Faulted	16028	66720	0.24
2FW04	90	Emergency / Faulted	11913	66720	0.18
2FW05	890	Thermal (Eqn. 10)	21822	22500	0.970
2FW07	A45/ B45	Sustained (Eqn. 8)	6540	15000	0.44
2MS01	225T	Emergency / Faulted	25039	39825	0.629
2MS02	110T	Emergency / Faulted	30149	39825	0.76
2MS03	45T	Emergency / Faulted	28207	39825	0.71
2MS04	155T	Emergency / Faulted	20644	39825	0.52
2MS05	23	Emergency / Faulted	10184	45479	0.22
2MS06	23	Emergency / Faulted	10184	45479	0.22
2MS07	23	Emergency / Faulted	9577	46479	0.21
2MS08	23	Emergency / Faulted	9463	45479	0.208
2MS25	80	Emergency / Faulted	19622	46125	0.43
2RH77K	A5	Emergency / Faulted	19214	39825	0.48

The maximum usage factors for each of the piping subsystems impacted by power uprate are provided below. All calculated usage factors satisfy the ASME Code requirements.

TABLE 3B-3: MAXIMUM USAGE FACTORS		
Subsystem	**Location**	**Maximum Usage Factor**
1FW01	A100	0.702
1FW02	A113	0.641
1FW03	95	0.131
1FW04	90	0.099
1MS01	225T	0.092
1MS02	180T	0.87
1MS03	45T	0.312
1MS04	155T	0.293
1MS05	15	0.0855
1MS06	15	0.081
1MS07	15	0.079
1MS08	15	0.083
1MS25	300	0.5282
1RH77K	1168	0.8139
2FW01	A139	0.640
2FW02	A113	0.990
2FW03	100	0.107
2FW04	90	0.099
2MS01	225T	0.09
2MS02	180T	0.865
2MS03	45T	0.312
2MS04	155T	0.293
2MS05	15	0.067
2MS06	15	0.086
2MS07	15	0.0795
2MS08	15	0.0831
2MS25	200	0.1771
2RH77K	A5	0.5795

For the subsystems identified above, all piping supports, penetrations, and anchors were evaluated for the impact of 105% power uprate. They all are within their allowable limits.

The Code of Record, Code allowables, and analytical techniques used in the power uprate evaluations are the same as those used in the original and existing design basis piping stress qualifications.

Neither new postulated pipe break nor new in-service inspection (ISI) weld locations were identified due to 105% Power Uprate.

Question 3c:

Provide similar information for the balance-of-plant piping systems evaluated as listed in Section 3.11.

Response 3c:

Please see responses to Questions 3a and 3b above for large bore balance of plant piping.

Methods similar to those discussed in the responses to questions 3a and 3b were used to evaluate the impact of power uprate on small bore balance of plant piping stress analyses. The power uprate parameters affected by power uprate (pressure, temperature, and flow) were compared to the parameters used in the existing design basis analyses. The increases in these parameters due to power uprate were evaluated for impact on the existing piping analysis results (pipe stress, support loads, anchor loads, etc.). In all cases, the increases in pressure, temperature, and flow were found to have only a small impact on piping stresses, support loads, anchor loads, etc. The small increases were concluded to have a negligible impact on the results of the existing piping stress analyses. The small increases are due to the conservative nature of the existing design basis calculations.

The Code of Record, Code allowables, and analytical techniques used in the power uprate evaluations for small bore piping are the same as those used in the original and existing design basis piping stress qualifications.

Question 4:

Provide an evaluation of the potential of flow induced vibration for the main steam and feedwater piping systems and for heat exchangers of the condensate and feedwater systems for the proposed power uprate.

Response 4:

Flow induced vibration for the main steam and feedwater piping systems was evaluated by extrapolating the data obtained from the LaSalle Unit 1 and Unit 2 startups, and comparing the extrapolation with the allowable vibration. It was concluded that the flow-induced vibrations will be within allowable limits. The increase in feedwater and extraction steam flows due to power uprate is small and is therefore not expected to have any significant impact on flow induced vibration of the heat exchangers. This conclusion is consistent with GE experience from similar power plant uprates. Evaluations of the effects of slightly increased steam flow from power uprate on the condensate and feedwater systems are ongoing, and it will be confirmed that there is no adverse impact.

Question 5:

Discuss the methodology and assumptions used for evaluating balance of plant (BOP) piping, components, and pipe supports, nozzles, penetrations, guides, valves, pumps, heat exchangers and anchors in Section 3.11 of the referenced topical report. Were the analytical computer codes used in the evaluation different from those used in the original design-basis analysis? If so, identify the new codes and provide justification for using the new codes and state how the codes were qualified for such applications.

Response 5:

The methodology used for BOP piping, component, and pipe support evaluations is the same as provided in the response to NRC Questions 3a and 3b. Piping interfaces with the nozzles, penetrations, guides, valves, pumps, heat exchangers and anchors in License Amendment Request Attachment E Section 3.11 were evaluated in a similar manner as provided in response to NRC Question 3a. In the evaluation process, the analytical computer codes used were the same as in the original design basis analysis.

Question 6:

Provide an evaluation of piping systems submerged in the suppression pool, vent penetrations, pumps, and valves, that may be affected by the LOCA dynamic loads (pool swell, condensation oscillation, and chugging) and the projected increase in the pool temperature as a result of the proposed power uprate.

Response 6:

The LaSalle power uprate does not include an increase in reactor operating pressure or SRV setpoints. Therefore, the existing LOCA load definitions bound the expected LOCA loads for power uprate conditions.

As shown in the License Amendment Request Attachment E Table 4-1, the calculated peak Design Basis Accident (DBA)-LOCA suppression pool temperature increases from a value of 190°F at current rated power to 193°F at the uprated power. The peak suppression pool temperatures calculated at current and uprated power are lower than the UFSAR value of 200°F due to the use of ANS 5.1-1979, "American National Standard for Decay Heat Power in Light Water Reactors," plus two sigma decay energy in the current suppression pool temperature analysis (see License Amendment Request Attachment E Section 4.1.1.1).

Because the LOCA dynamic loads (pool swell, condensation oscillation, and chugging) are within their design basis and the peak suppression pool temperature is bounded by the UFSAR value, there is no effect due to power uprate on the design basis for the piping systems submerged in the suppression pool, or for the vent penetrations, pumps and valves.

Question 7:

Do you project modifications to piping or equipment supports for the proposed power uprate? If so, provide examples of pipe supports requiring modification and discuss the nature of these modifications.

Response 7:

No modifications to piping or supports are required due to power uprate implementation.

Question 8:

As a result of plant operations at the proposed uprated power level, the decay heat load for any specific fuel discharge scenario will increase. Please provide the following information:

a. Provide the heat loads and corresponding peak calculated spent fuel pool (SFP) temperatures during planned refueling outages[1] and unplanned full core off-load.

[1] If full core off-load is the normal practice during planned refueling outages at LaSalle, a single failure of the SFP cooling system should be assumed in the SFP thermal analysis for the planned refueling outages. A single failure of the SFP cooling system need not be assumed for the unplanned full core off-load events.

b. Since the residual heat removal (RHR) system serves as a back-up system to the SFP cooling system, prior to a planned or unplanned full core offload event, how many trains of SFP cooling system and RHR system are required to be operable and available for SFP cooling?

c. Discuss the provisions that have been established in the plant operating procedures to ensure that the RHR system will be aligned for SFP cooling.

Question 8a:

Provide the heat loads and corresponding peak calculated spent fuel pool (SFP) temperatures during planned refueling outages[1] and unplanned full core off-load.

Response 8a:

In order to conform to the guidance presented in NRC Standard Review Plan (SRP) 9.1.3, "Spent Ruel Pool Cooling and Cleanup System". The following conservative and bounding fuel discharge scenarios were analyzed for the power uprate Fuel Pool Cooling and Cleanup System (FPCCS) assessments:

First, a spent fuel pool (SFP) is assumed to be almost-filled with spent fuel assemblies (SFAs) from previous refuelings, such that there are only spaces available for one more batch offload, plus one full core offload. A batch (i.e., "normal") offload into the pool is assumed to proceed during a refueling outage lasting 20 days. This is Case 1. Case 1 analysis results provide the "maximum normal heat load," to use the SRP 9.1.3 terminology.

The plant then returns to power, and full-power operation is assumed for 36 days (this 36 day period corresponds to the time assumed in SRP 9.1.3, Section III.1.h.iii). At that point, an "emergency" arises that requires the offload of the full core into the SFP (which contains the batch previously offloaded plus the previous inventory of SFAs). This is Case 2. Case 2 analysis results provide the "abnormal maximum heat load," again using the SRP 9.1.3 terminology.

For both Case 1 (batch) and Case 2 (core) offload scenarios, the following facts are noted:

- Regarding the footnote in NRC Question 8a, for future refueling outages LaSalle plans to offload only a "batch" of fuel, versus the previous practice of offloading a full core. This allows a shorter outage duration (although this does not preclude an occasional full core offload if circumstances such as cycle-specific fuel considerations require one). For purposes of the bounding power uprate SFP analyses, a batch is assumed to consist of 320 SFAs, and the offload rate is assumed to be an average rate of 15 SFAs per hour. A full core consists of 764 SFAs. This applies to both LaSalle Units.

- The LaSalle SFP capacities are 4078 cells in the Unit 2 SFP, and a Unit 1 SFP capacity of 3986 cells (including fuel stored in the defective fuel storage locations). For a bounding analysis, the Unit 2 SFP is analyzed, assuming an initial inventory of (4078 - 764 - 320) 2994 cells filled with previous offloads. The Unit 2 SFP capacity bounds the Unit 1 SFP. The previous batches are assumed to have been irradiated at the power uprate power level, 3489 MW_t, at a 97% capacity factor, for up to three 24-month fuel cycles. Decay heat is calculated using the ANS 5.1-1979 Standard with two sigma uncertainty. (See the response to NRC Question 11.0 for a discussion of the decay heat calculations compared to BTP ASB 9-2).

- The time before fuel offload begins is assumed to be 24 hours. This assumption is based on Technical Specification 3.9.4, which requires the reactor to be subcritical for at least 24 hours prior to movement of irradiated fuel in the reactor pressure vessel.

- For one batch offloaded into the SFP ("maximum normal heat load"), in accordance with single failure considerations, the FPCCS is assumed to be limited to only one pump (3000 gpm) through one heat exchanger (HX). An initial Service Water System temperature of 100°F is assumed, and the SFP is assumed to initially be at 100°F for these conditions. (The 100°F Service Water System value bounds a worst-case lake temperature; the 100°F SFP temperature correlates with an administrative control on SFP temperature during normal operation). This results in zero heat removal at the onset of the analysis.

- For the full core offloaded into the SFP ("abnormal maximum heat load"), the FPCCS capacity flowrate without a single failure is two pumps (5050 gpm total) through two heat exchangers. An initial Service Water System temperature of 100°F is assumed, with the SFP initially at 100°F. The 5050 gpm is based on the FPCCS capacity calculations with two pumps running.

The results of the power uprate SFP analyses, using the above assumptions, are as follows:

Case 1 (Maximum Normal Heat Load) (one batch of 320 SFAs):

As noted above, the SFP is assumed to be almost filled with previous refuel batches. The decay heat load from these previous batches is calculated to be 4.2 x 10^6 BTU/hr. The decay heat load into the SFP for the batch offload (320 SFAs) is calculated to be 19.8 x 10^6 BTU/hr. The design heat removal rating of a single spent fuel pool heat exchanger is 14.5 x 10^6 BTU/hr. Therefore, the spent fuel pool temperature will increase as the fuel assemblies are offloaded, peak at some temperature, then gradually decay as the heat exchanger dissipates the excess heat and its heat removal efficiency improves due to the difference in temperatures between the Service Water and the fuel pool water.

Initial Case:

For a batch offload commencing 24 hours after reactor shutdown, the SFP temperature peaks at 149°F at 54 hours (2.25 days) after reactor shutdown, then falls below 140°F at 116.4 hours (about 4.8 days) after reactor shutdown and continues to fall off. This is considered acceptable for the following reasons:

- This initial case was run as a bounding case, i.e., the analytical parameters were chosen such that they would never be coincidental or exceeded during actual plant operations. Thus the 140°F criterion was expected to be exceeded for this case;
- ANSI N210-1976/ANS-57.2, "Design Objectives for Light Water Reactor Spent Fuel Storage Facilities at Nuclear Power Stations," contains a design requirement for the peak SFP temperature of 150°F, with all cells filled;
- The SFP temperature does not exceed 150°F, and only exceeds the 140°F criterion for about 2½ days, then falls to 130°F or below within a week.

Case 1:

In order to maintain the peak SFP temperature at or below the SRP criterion of 140°F, iterative calculations were performed for various "in-reactor" hold times. If the offload of fuel is delayed until 74 hours after reactor shutdown, the SFP temperature will peak at 140°F at 105 hours (about 4.5 days) after reactor shutdown, then decay. See Figure 8a-1 below. (Note that, because of the initial assumptions of 100°F SFP water in conjunction with 100°F Service Water, the SFP temperature rises slightly prior to the offload, due to the residual decay heat in the pool from the previous pool inventory of SFAs). The UFSAR will be revised, as a result of these power uprate analyses, to require refueling-specific analyses under the current plant conditions to be conducted prior to offload, to ensure that the 140°F criterion is not exceeded during an offload commencing within 24 to 74 hours after shutdown. Also, see the Response to NRC Question 10.

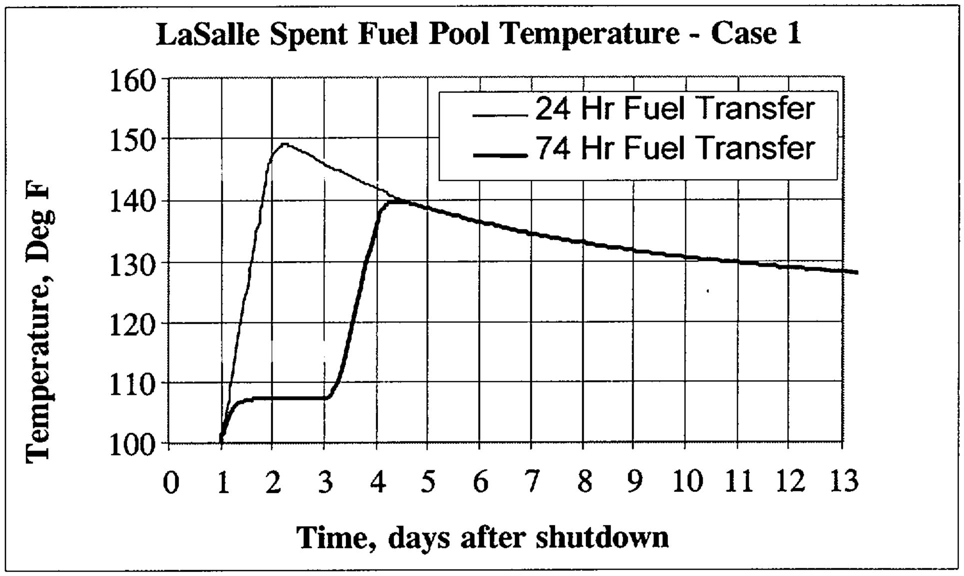

FIGURE 8a-1: Maximum Normal Heat Load: SFP Temperature Versus Days After Shutdown

<u>Case 2</u> (Abnormal Maximum or "Emergency" Heat Load) (one core offloaded into a full pool):

The previous Case 1 resulted in a SFP with 764 cells empty, or just enough for a full core "emergency" offload. Using the assumptions described above, after 20 days for the previous refueling, plus 36 days at power, plus 1 day (24 hours) following reactor shutdown, a full core is assumed to be offloaded into the SFP at 15 SFAs per hour. The SFP pre-existing heat load, including the 320 SFAs recently offloaded, is calculated to be 10.3 x 10^6 BTU/hr. The decay heat load into the SFP for the core offload (764 SFAs) is calculated to be 44.1 x 10^6 BTU/hr. The design heat removal rating using both spent fuel pool heat exchangers is calculated to be 26.53 x 10^6 BTU/hr (with a dual pump flowrate of 5050 gpm). Therefore, the spent fuel pool temperature will increase as the fuel assemblies are offloaded, peak at some temperature, then gradually decay as the heat exchangers dissipate the excess heat and their heat removal efficiency improves due to the difference in temperatures between the Service Water and the fuel pool water.

The full core offloaded into the SFP after 36 days operation following the previous batch offload results in a peak SFP temperature of 150.5°F at 78 hours after shutdown, assuming offload commences at 24 hours after shutdown. This is well below boiling, which is the acceptance criterion for this Abnormal Maximum Heat Load scenario. See Figure 8a-2 below.

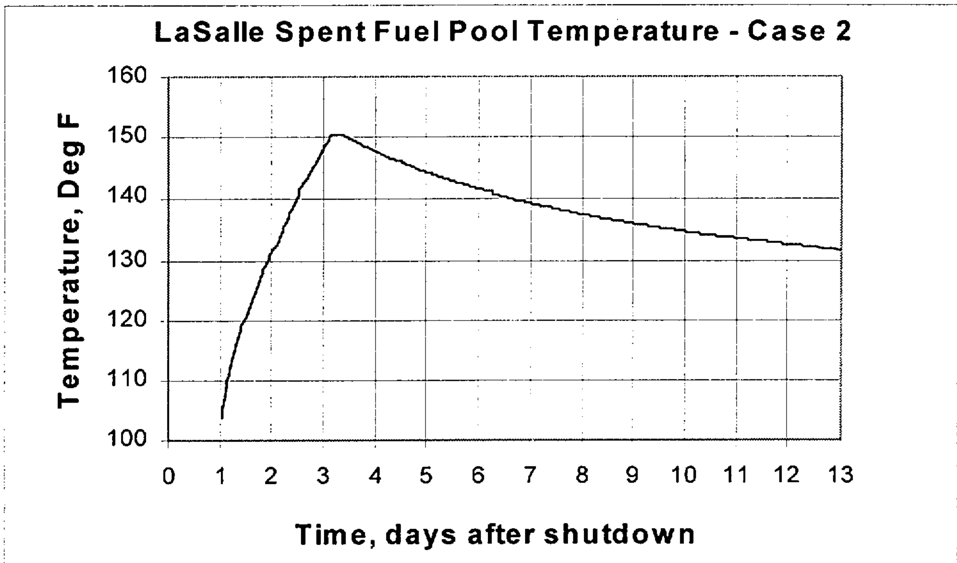

FIGURE 8a-2: Abnormal Maximum Heat Load: SFP Temperature Versus Days after Shutdown

Question 8b:

Since the residual heat removal (RHR) system serves as a back-up system to the SFP cooling system, prior to a planned or unplanned full core offload event, how many trains of SFP cooling system and RHR system are required to be operable and available for SFP cooling?

Response 8b:

The ComEd Shutdown Safety Management Program goal for every outage is that both trains of the FPCCS, or equivalent, are available during the refueling outage (except for scheduled electrical bus outages). For each refueling outage, Engineering determines on a cycle-specific and offload-specific basis, and with the current plant conditions, the SFP heatup rate and the time to boil if FPCCS should be lost. Note also that the opposite Unit's FPCCS can be used to provide cooling to the SFP by removing the Transfer Canal Gates/Cask Well Gates, and this contingency is described in the Shutdown Safety Management Program, and LaSalle FPCCS procedures, which note the following primary sources of fuel pool cooling:

- 1A Fuel Pool Cooling pump and heat exchanger available
- 1B Fuel Pool Cooling pump and heat exchanger available
- B RHR pump and heat exchanger available for Fuel Pool Cooling assist
- 2A Fuel Pool Cooling pump and heat exchanger available
- 2B Fuel Pool Cooling pump and heat exchanger available

The above measures ensure that adequate cooling capability is provided for planned refuelings, either a core shuffle or a full core offload, and for unplanned offloads. Note that there is no requirement that the RHR system be operable and available for SFP cooling during an offload; also see the response to Question 8c.

As noted in Response 8a, LaSalle does not envision future full-core offloads as a normal refueling evolution; rather, a core shuffle, (i.e., partial-core, or "batch") offload is planned for future outages, although this does not preclude an occasional full core offload if circumstances require it. Therefore, there may be no "planned" full core offloads; an "unplanned" full core offload would be governed by the same requirements as for a "planned" offload, which are the existing Technical Specification (TS) requirements and existing procedural controls as described herein.

The LaSalle "Refueling Operations" Technical Specifications are found at TS Sections 3/4.9:

- TS 3.9.8, "Water Level-Reactor Vessel", requires that 22 feet of water be maintained over the top of the reactor pressure vessel flange during handling of SFAs or control rods within the vessel when the SFAs are being handled, or when the SFAs seated in the vessel are irradiated.

- TS 3.9.9, "Water Level-Spent Fuel Storage Pool", requires that 23 feet of water be maintained over the top of active fuel in SFAs seated in the SFP racks whenever irradiated fuel assemblies are in the SFP.

- TS Sections 3/4.9.11, "Residual Heat Removal and Coolant Circulation," contains the following requirements for the availability of the RHR System when the plant is in Operational Condition 5, Refueling.

    - TS 3.9.11.1 requires at least one shutdown cooling mode loop of the RHR system be OPERABLE and in operation when irradiated fuel is in the reactor vessel and the water level is ≥ 22 feet above the top of the reactor pressure vessel flange.

- TS 3.9.11.2 requires two shutdown cooling mode loops of the RHR system be OPERABLE and at least one loop in operation when irradiated fuel is in the reactor vessel and the water level is < 22 feet above the top of the reactor pressure vessel flange.

TS 3.7.1.1 "Core Standby Cooling System - Equipment Cooling Water Systems, Residual Heat Removal Service Water", requires two independent RHR Service Water System subsystems be operable in Operational Conditions 1 through 5 (with only one pump required, if sufficient for decay heat removal, in Operating Conditions 4 and 5).

<u>Question 8c:</u>

Discuss the provisions that have been established in the plant operating procedures to ensure that the RHR system will be aligned for SFP cooling.

<u>Response 8c:</u>

A LaSalle Procedure is in place to institute the RHR Fuel Pool Cooling Assist Mode, if required, on loss of both Units' FPCCS. Note that there is no requirement that the RHR system be aligned for SFP cooling during an offload; rather, the RHR System is designed to be available by spooling in suction and return lines to the RHR "B" loop pump and heat exchanger of that unit, to provide backup cooling capability to the SFP if an "emergency" condition requires this. See UFSAR 9.1.3.2.1, "Fuel Pool Cooling." The NRC reviewed and approved this design during the Operating License (OL) phase, as noted in the OL SER (NUREG-0519, March 1981) Section 9.1.3 which states in part:

> "The residual heat removal pumps can be cross connected to the fuel pool cooling system to use the residual heat removal system to cool the spent fuel, if necessary."
>
> ***
>
> "Based on our review, we conclude that the design of the spent fuel pool cooling and cleanup system is in conformance with the requirements of Criteria 61 and 62 of the General Design Criteria, Branch Technical Position ASB 9-2 with respect to decay heat loads, and the positions in

Regulatory Guides 1.13 and 1.29, including the positions on availability of assured makeup sources but excluding the position on seismic Category I classification which is justified by dose release analysis described above and is, therefore, acceptable."

Refer to the responses to Question 8b above for information regarding the Technical Specification requirements for RHR system operability during Operational Condition 5, Refueling.

Question 9:

(a) As stated in the Updated Final Safety Analysis Report (UFSAR), the SFP cooling system is designed to maintain the SFP at or below 120°F with a decay heat load of 14.5 x 10^6 Btu/hr from all the previously discharged spent fuel assemblies (SFAs) and a freshly discharged partial (1/3) core. *(b)* In the power uprate submittal, the SFP temperature is allowed to rise to 140°F during planned (normal) refueling outage. *(c)* Also, as stated in the UFSAR, in an event of an unplanned (emergency) full core offload, the SFP temperature will be maintained below 150°F. *(d)* In the power uprate submittal, ComEd stated that the SFP temperature is allowed to rise to below pool boiling. *(e)* Discuss the effects of these elevated pool temperatures during planned and unplanned full core off-load events on SFP (i.e. structures, SFP linings, etc.) and the SFP cooling and cleaning systems.

Response 9:

As shown above, NRC Question 9 is broken into five parts (designated by italic letters), and our responses separately address each part to provide additional clarification.

Question 9*(a)*:

As stated in the Updated Final Safety Analysis Report (UFSAR), the SFP cooling system is designed to maintain the SFP at or below 120°F with a decay heat load of 14.5 x 10^6 Btu/hr from all the previously discharged spent fuel assemblies (SFAs) and a freshly discharged partial (1/3) core.

RESPONSE 9*(a)*:

The design of the FPCCS is to maintain the SFP at or below 120°F during normal operation. The FPCCS originally was designed to maintain the SFP at or below 120°F during a refueling batch offload, consisting of 1/3 of a core irradiated for 3 years and decayed for 150 hours, into the SFP which contained 1/3 of a core which had been irradiated for 3 years and had been in the SFP for 1 year (UFSAR 9.1.3.1.1).

However, during the Unit 1 and 2 reracks (Unit 2 Amendment 48 to Facility Operating License No. NPF-18, June 15, 1989; and Unit 1 Amendment 90 to Facility Operating License No. NPF-11, February 24, 1993), the SFP inventory basis changed (from a 1 and 1/3 core capability, to over 5 cores), as did the refueling intervals assumed (from 12 month cycles to 18 month cycles). The Unit 2 rerack submittal analyzed a batch offload of 240 SFAs 7 days (168 hours) after reactor shutdown, into the SFP which contained 2880 previous SFAs. One train of FPCCS was assumed to be operating. The offload was assumed to occur at the rate of only 4 SFAs per hour. Under these conditions, the SFP temperature was calculated to reach a peak temperature of 119.6°F. (The Unit 1 rerack submittal did not analyze a batch offload, or a batch offload peak SFP temperature).

Question 9*(b)*:

In the power uprate submittal, the SFP temperature is allowed to rise to 140°F during planned (normal) refueling outage.

Response 9*(b)*:

The power uprate analyses are being performed to bound all previous cases, and any future batch offloads, by assuming that 320 SFAs are offloaded into an SFP containing 2994 SFAs. In addition, the power uprate analyses assume irradiation of all SFAs at the uprated core thermal power (3489 MWt), 24-month refueling cycles, a Service Water temperature of 100°F, and an initial SFP temperature of 100°F. The time after reactor shutdown is evaluated for 24 hours (with a peak SFP temperature of 149°F) and for 74 hours (with a peak SFP temperature of 140°F). The UFSAR will be revised to clarify the 140°F limit (i.e., the SRP 9.1.3 acceptance criterion) for a batch offload.

Question 9*(c)*:

Also, as stated in the UFSAR, in an event of an unplanned (emergency) full core offload, the SFP temperature will be maintained below 150°F.

Response 9*(c)*:

The FPCCS originally was designed to maintain the SFP at or below 150°F during an "emergency" full core offload. Note that this offload, consisting of a core irradiated for 3 years and decayed for 150 hours, was into an SFP which contained only 1/3 of a core which had been irradiated for 3 years and had been in the SFP for 100 days (UFSAR 9.1.3.1.1).

However, during the Unit 1 and 2 reracks (see response to Question 9a), the SFP capacities changed (from a 1 and 1/3 core capability, to over 5 cores), as did the refueling interval assumed (from 12 month cycles to 18 month cycles). The assumptions for SFP inventory, time at power before the full core offload, and the time after reactor shutdown were also revised from the original UFSAR assumptions.

The first (Unit 2) rerack submittal analyzed a full core offload 7 days (168 hours) after reactor shutdown, into the SFP, which contained 3120 previous SFAs. The reactor was assumed to be operating for 30 days after the previous batch offload had been placed in the SFP. A refueling outage length (i.e., additional decay time for the SFP inventory) was not specified. Only 1 train of FPCCS was assumed to be operating. Under these conditions, the SFP temperature was calculated to reach a peak temperature of 149.4°F. The NRC used a higher decay heat load, and calculated that the SFP reached a peak temperature of 168°F. This SFP temperature was acceptable to the NRC (Amendment 48 to Facility Operating License No. NPF-18, June 15, 1989), based on the SRP 9.1.3 criterion of no boiling.

The next (Unit 1) rerack submittal analyzed a full core offload 100 hours after reactor shutdown, into the SFP, which contained 3328 previous SFAs. The reactor was assumed to be operating for 30 days after the previous batch offload had been placed in the SFP. A refueling outage length of 45 days (i.e., additional decay time for the SFP inventory) was specified. Both trains of FPCCS were assumed to be operating. Under these conditions, the SFP temperature was calculated to reach a peak temperature of 127.9°F. The SRP 9.1.3 criterion of no boiling was met and was the basis of acceptance by the NRC (Amendment 90 to Facility Operating License No. NPF-11, February 24, 1993). Also refer to the discussions in Response 8a.

Question 9*(d)*:

In the power uprate submittal, ComEd stated that the SFP temperature is allowed to rise to below pool boiling.

Response 9*(d)*:

The intent of the power uprate submittal and related SFP analyses is to provide closer correlation to the SRP 9.1.3 guidance. As seen by the discussions provided in the response to Question 9c, the historical UFSAR terminology and analyses can be confusing when compared to the rerack submittal information. As documented in the power uprate submittal, the SRP 9.1.3 acceptance criteria of 140°F for a batch offload, and of no pool boiling for an "emergency" full core offload, are the power uprate acceptance criteria (and were the Staff's acceptance criteria for the rerack submittals). For an "emergency" full core offload, the acceptance criterion for the SFP temperature is that no boiling occur, assuming that both trains of FPCCS are operating. As shown in our Response 8a, the SFP temperature reaches a peak temperature of 150.5°F for the "emergency" full core offload case analyzed (Case 2), whereby a full core is offloaded 36 days following the batch offload during a 20-day refueling outage. Since the UFSAR must be updated to reflect the power uprate SFP analyses, the UFSAR will be clarified on these parameters discussed in our responses. (As another example, the UFSAR "Maximum Normal Heat Load (MNHL)" and "Emergency Heat Load (EHL)" bases are modified by both the rerack submittals and by the power uprate analyses, and this will be clarified also in the planned UFSAR update).

Question 9*(e)*:

Discuss the effects of these elevated pool temperatures during planned and unplanned full core off-load events on SFP (i.e. structures, SFP linings, etc.) and the SFP cooling and cleaning systems.

Response 9*(e)*:

The "elevated" SFP temperatures during planned and unplanned events with one or both FPCCS trains operating, as shown by the analysis results in Response 10a, Table 10a-1 below, do not exceed 151.3°F for a full core offloaded into a "full" SFP after a 24-month fuel cycle (Case 2a). The SFP, including the liner and the reracks, has been analyzed for normal operation at temperatures ranging to 140°F. The SFP, including the liner and the reracks, has been analyzed for an accident condition temperature of 212°F. The reinforced concrete structural components have also been analyzed for an accident SFP temperature of 212°F. The FPCCS piping, pumps and heat exchangers are not affected by operation at 150°F. The piping, pumps, and heat exchangers are not affected by operation slightly above 150°F (i.e., 150.5°F for Case 2 or 151.3°F for Case 2A) for the times calculated for the "emergency" offload scenarios. The maximum operating temperature of 151.3°F has a negligible impact on piping stress analyses and support/equipment qualifications. Non-metallic materials of construction for the FPCCS pumps and heat exchangers are not affected by SFP water temperatures up to 155°F.

The filter/demineralizers are completely bypassed if the inlet water temperature should exceed 150°F (UFSAR 9.1.3.2).

Question 10:

Since the heat removal capability of the SFP cooling system is a function of the lake temperature and the decay heat load is a function of the SFAs "in-reactor" hold time prior to discharge SFAs from the reactor, ComEd stated in the UFSAR that:

> "Normal refueling outages are planned to control the start of core offloading and/or the time of year (i.e., expected lake water temperature) in which the outage will occur."

Please provide the following information:

a. The calculated SFP peak temperatures at various lake water temperatures (i.e. 40°F, 60°F, 80°F 90°F, 95°F, etc.) and their corresponding SFAs "in-reactor" hold time required; coincident[2] time after reactor shutdown; and coincident decay heat load. For the case with the highest decay heat load, also provide the "time-to-boil" and maximum boil off rate.

b. Discuss the provisions established or to be established in plant operating procedures that require analyses to be performed to determine SFAs "in-reactor" hold time to ensure that the SFP operating temperature limit of 140°F will not be exceeded.

Question 10a:

The calculated SFP peak temperatures at various lake water temperatures (i.e., 40°F, 60°F, 80°F 90°F, 95°F, etc.) and their corresponding SFAs "in-reactor" hold time required; coincident[2] time after reactor shutdown; and coincident decay heat load. For the case with the highest decay heat load, also provide the "time-to-boil" and maximum boil off rate.

---

2 The time after reactor shutdown at which the SFP water reaches its temperature limit of 140°F.

Response 10a:

For a batch or core offload from the reactor vessel, the heat removal capability of the FPCCS is dependent on the previous inventory (decay heat load) in the SFP, the SFP shell side temperature, the Service Water tube side inlet temperature, the irradiation history of the batch or core to be offloaded, and the time before the offload commences.

ComEd's Nuclear Generation Group priority is to keep all ten nuclear units online all summer in order to accommodate the peak summer electrical loads experienced. Therefore, there are no refueling outages scheduled for LaSalle during the summer, when the lake temperatures are high. However, in order to minimize the heat removal capability of the FPCCS, the previous UFSAR, rerack, and power uprate offload analyses assume a Service Water temperature much higher than that seen in the actual Fall refueling outages. This assumption maximizes, and bounds, the SFP temperature for the cases evaluated. Therefore, the bounding power uprate cases are presented herein, for a Service Water temperature of 100°F, rather than iterations on lower Service Water temperature. Other parameters and assumptions that maximize the calculated SFP temperature include:

- SFP temperature of 100°F
- SFP filled with 2994 SFAs assumed irradiated at 3489 MWt, with 24-month fuel cycles
- Fuel being offloaded irradiated at 3489 MWt for 24-month fuel cycles at a 97% capacity factor
- 320 SFAs are offloaded for the batch offload cases
- 24-hour offload after reactor shutdown ("in-reactor" hold time) core offload, plus 74-hour time for the batch offload (batch also analyzed for 24-hour offload)
- 15 SFAs transferred per hour
- 20-day refueling outage

The results of the power uprate SFP analyses, using the above assumptions, are depicted in Table 10a-1 below.

An "in-reactor" hold time of 24 hours before a batch offload (initial case) gives a calculated peak SFP temperature of 149°F using the above assumptions, whereas delaying the batch offload for 74 hours (Case 1) gives a calculated peak SFP temperature of 140°F. The case with the highest decay heat load is Case 2a, whereby a full core is offloaded into the SFP, which contains the previous inventories of uprated SFAs, and the batch offloaded in the last refueling. The highest peak decay heat load for this Case 2a is 55.4 x 10^6 BTU/hr, assuming the core is offloaded 24 hours after shutdown. Also see the response to Question 10b below.

The times to boil, and the boil off rates, for Cases 1 through 2a are as shown in Table 10a-1.

As pointed out previously, the assumptions (depicted in Table 10a-1) are all chosen to be conservative, in order to bound any future offloads.

In reality, the SFP heat loads are expected to be much lower than those calculated. For example, in April 1999, the following SFP parameters were calculated:

Parameter	Unit 1 SFP as of 04/14/99	Unit 2 SFP as of 04/14/99
Total Decay Heat Load (BTU/hr)	1.18 E06	1.44 E06
Temperature Rise if FPCCS is lost (Degrees F/hr)	0.36	0.45
Time To Boil if FPCCS is lost (hours)	322	263

TABLE 10A-1 SPENT FUEL POOL DECAY HEAT LOAD RESULTS

Power Uprate SFP Case	Peak Heat Load (which then decreases with time)	Time to Boil (from peak Temp.)	Boil off Rate (at 212°F)	Peak SFP Temp. (with FPCC), at time after shutdown	Remarks
Initial Case: Batch offload of 320 SFAs 24 hours after shutdown (S/D)	25.6 X 10^6 BTU/hr (plus 4.2 X 10^6 in SFP at start) Total 29.8 X 10^6 BTU/hr	Not calculated	Not calculated	149°F at 54 hours after S/D (> 140°F for 62.4 hours)	Pool almost full (2994 cells filled) w/ uprate SFAs. After offload, the SFP contains 3314 SFAs. 1 FPCC pump and HX operating.
Case 1: Batch offload of 320 SFAs 74 hours after S/D	19.8 X 10^6 BTU/hr (plus 4.2 X 10^6 in SFP at start) Total 24.0 X 10^6 BTU/hr	8.4 hrs	47 gpm	140°F at 105 hrs after S/D	Pool almost full (2994 cells filled) w/ uprate SFAs. After offload, the SFP contains 3314 SFAs. 1 FPCC pump and HX operating.
Case 2: Full core offload of 764 SFAs 24 hours after S/D.	44.1 X 10^6 BTU/hr (plus 10.3 X 10^6 in SFP at start) Total 54.4 X 10^6 BTU/hr	2.9 hrs	110 gpm	150.5°F at 78 hrs after S/D	36-day operation after Case 1 batch offload and 20-day refuel. After offload, the SFP contains 4078 SFAs. 2 FPCC pumps and HXs operating.
Case 2a: Full core offload of 764 SFAs 24 hours after S/D.	51.0 X 10^6 BTU/hr (plus 4.4 X 10^6 in SFP at start) Total 55.4 X 10^6 BTU/hr	2.8 hrs	112 gpm	151.3°F at 78 hrs after S/D	24-month operation after Case 1 batch offload and 20-day refuel. After offload, the SFP contains 4078 SFAs. 2 FPCC pumps and HXs operating.

Question 10b:

Discuss the provisions established or to be established in plant operating procedures that require analyses to be performed to determine SFAs "in-reactor" hold time to ensure that the SFP operating temperature limit of 140°F will not be exceeded.

Response 10b:

The LaSalle Station Fuel Handling Surveillance Procedure requires that the SFP can be maintained below 140°F with only one Fuel Pool Cooling and Cleanup System in operation during fuel moves from the reactor to the pool. This is applicable to a "normal" refueling operation, which as noted above can include a batch offload or a full core offload. The same procedure requires calculation of the cycle-specific, offload-specific SFP decay heat load, maximum allowed heat load in the SFP assuming only one train of FPCCS is operating, and the time delay (i.e., "in-reactor hold time") before core offload is allowed. As noted in Response 10a, these actual heat loads are expected to be much less than those calculated in the power uprate SFP analyses. LaSalle procedures control the calculation of decay heat loads and heat removal capabilities of the redundant and diverse means of achieving SFP cooling prior to and during fuel movements.

Question 11:

With regard to the SFP cooling, is the Branch Technical Position ASB 9-2, "Residual Decay Energy for Light Water Reactors for Long-Term Cooling" used to perform the decay heat calculations? If not, identify and provide the basis for any deviations and exceptions to the guidance described in Section 9.1.3 of the Standard Review Plan (SRP, NUREG-0800) including the Branch Technical Position ASB 9-2. Also, discuss the assumptions used to calculate the decay heat.

Response 11:

The GE power uprate evaluation is based on the ANSI/ANS-5.1-1979 decay heat standard, with the addition of two sigma uncertainty. This Standard was derived from a combination of experimental data and calculations. These power fractions are shown (Figure 11-1) to be representative of the curves presented in the NRC Branch Technical Position (BTP) ASB 9-2.

The NRC Standard Review Plan 9.1.3 recommends using the decay heat presented in the NRC Branch Technical Position, ASB 9-2. The decay heat curve used for these evaluations, ANS 5.1-1979 Standard with two sigma, is compared to the decay heat curve in ASB 9-2, in Figure 11-1. The decay curve shown in ASB 9-2 uses lower exposure than this GENE application, and therefore the ASB 9-2 decay curve was adjusted (increased) to match the exposure used in this application. Throughout the entire application range of 1 day to 20 years (8.64 E4 seconds to 6.3 E8 seconds), the ANS 5.1-1979 standard with two sigma has a higher decay heat ratio than the ASB 9-2 curve except for the following ranges:

- The ANS 5.1-1979 Standard decay heat is less than the ASB 9-2 decay heat (by less than 1%) between 15 days (1.3 E6 seconds) and 30 days (2.59 E6 seconds). The SFP peak pool temperature occurs much earlier than 15 days for the cases analyzed.

- The ASB 9-2 curve is higher than ANS 5.1-1979 Standard beyond 12 years (3.8 E8 seconds). This is more than compensated for in the calculation of the decay heat load from the previous cycles, by the substantially higher ANS 5.1-1979 Standard in the time period between 30 days and 12 years. In addition, the decay heat in this extended time range is so low that it does not affect the heat load calculation in any significant way.

Overall comparison of the two decay heat standards shows that the decay heat calculated by the ANS 5.1-1979 Standard is higher than that calculated by the ASB 9-2 curve for the SFP temperature calculations for LaSalle. This is true for both the heat loads resulting from the fuel discharged for Cases 1, 2, and 2a and for the heat loads from the previous cycles. Therefore, it is conservative to use the ANS 5.1-1979 Standard instead of the ASB 9-2 decay curve.

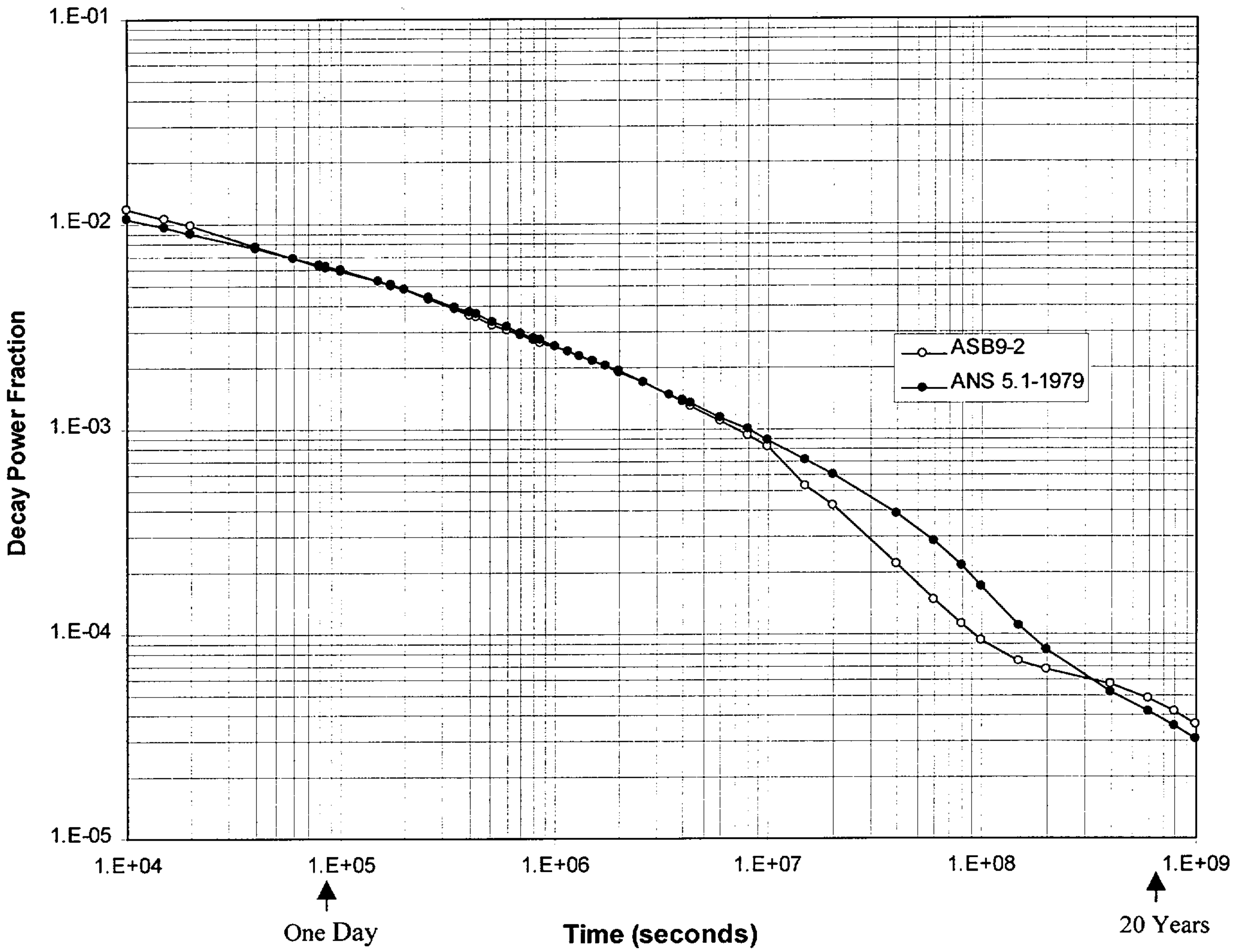

Figure 11-1: Decay Heat Curve Comparison Between ANS 5.1-1979 and ASB 9-2

The following additional assumptions were used in calculating decay heat loads:

- The uprated power of 3489 MWt was used.
- Fuel bundle average enrichment of 4.25% was assumed with GE12 fuel. Enrichment, exposure, and other important properties are bounded by the values used in this evaluation.
- The reactor operating cycle of 24 months was assumed with a capacity factor of 97%.

The calculated decay heat values were based on the total decay heat load for 24-month fuel cycles and core average exposure of 36,000 MWd/MT with a capacity factor of 97%, and with 5% power uprate for GE-12 equilibrium fuel cycle. The impact of the power uprate was determined by calculating decay heat values for power uprate for applicable Cases 1 and 2.

The ANS curves calculated were for 24-month cycles with the higher core average enrichments reflective of power uprate. The power uprate curves include a higher average enrichment, a higher exposure and a longer irradiation time.

<u>Case 1 Assumptions:</u>

For the 320 bundles discharged, an exposure of 56 GWd/MT with 6 years of uprate power operation was used. The heat load from the previous uprate cycles is assumed to be from the 2994 fuel bundles discharged in the previous cycles, with an exposure of 56 GWd/MT for all batches, and batches have been in the SFP for the following duration for each batch:

114 bundles	20 years
320 bundles	18 years
320 bundles	16 years
320 bundles	14 years
320 bundles	12 years
320 bundles	10 years
320 bundles	8 years
320 bundles	6 years
320 bundles	4 years
320 bundles	2 years

<u>Case 2 Assumptions:</u>

The decay heat for Case 2 is based on the full-core discharge of 764 bundles after 36 days of power operation after the last refueling outage. The full core consists of the following three groups of fuel bundles:

- 320 bundles 36.88 GWd/MT of exposure with 4 years and 36 days in power operation
- 320 bundles 18.88 GWd/MT of exposure with 2 years and 36 days in power operation
- 124 bundles 0.88 GWd/MT of exposure with 36 days in power operation

The decay heat load in the SFP from the previous cycles is the same as Case 1 except that the bundles are decayed 56 days longer and there is one additional batch of 320 bundles that is decayed only 56 days.

Case 2A Assumptions:

The decay heat for Case 2A is the same as for Case 2 except it is assumed that the reactor has continued full power operation to the end of the full cycle of two years. As is customary in full core decay heat load calculations, the core was conservatively assumed to consist of 764 bundles at the average exposure:

- 764 bundles 36 GWd/MT of exposure with 4 years in power operation.

The decay heat load in the SFP from the previous cycles is the same as Case 1 except that they are decayed 2 years longer and there is one additional batch of 320 bundles that is decayed for 2 years.

Question 12:

In the unlikely event that there is a complete loss of SFP cooling capability, the SFP water temperature will rise and eventually will reach boiling temperature. Provide the time to boil (from the pool high temperature alarm caused by loss-of-pool cooling to boiling) and the boil-off rate (based on the highest heat load from the planned or unplanned full core off-load). Also, discuss sources and capacity of make-up water and the methods/systems (indicating system seismic design Category) used to provide the make-up water.

The above information is necessary to allow the staff to determine whether the analyses are consistent with the guidance described in Standard Review Plan, Section 9.1.3, "Spent Fuel Pool Cooling and Cleanup System."

Response 12:

The peak heat loads, peak SFP temperature, time of peak SFP temperature, time to boil assuming loss of all FPCCS cooling, and the boil off rates for the cases analyzed for power uprate, are depicted in the previous response to NRC Question 10a above (see Table 10a-1). The time to boil was calculated by assuming that FPCCS cooling is lost at the time of peak SFP temperature for each case.

The FPCCS conforms to the guidance in NRC SRP 9.1.3, including a seismic Category I makeup system and an appropriate backup method to add coolant to the SFP. The review procedures of SRP 9.1.3 section III.1.f state that the backup system need not be a permanently installed system, nor Category I, but must take water from a Category I source. As stated in UFSAR section 9.1.3.3, a seismic Category I fuel pool emergency makeup system, which is capable of a makeup rate of 300 gpm to the pool, is present should it be necessary to provide emergency makeup. The emergency fuel pool makeup system is part of the core standby cooling system-equipment cooling water system (CSCS-ECWS) which is also seismic Category I, thus ensuring a reliable source of water. Makeup to the pool is provided by either of two redundant subsystems in separate electrical divisions, thus ensuring an adequate means of maintaining the fuel pool level.

The NRC reviews and approvals of the FPCCS, including the emergency makeup system, are documented as discussed below.

The FPCCS analyses are consistent with the guidance in NRC SRP 9.1.3. The original OL NRC reviews of the Spent Fuel Cooling and Cleanup Systems (NRC Safety Evaluation Report, NUREG-0519, March 1981) are described, in part, as follows:

> "9.1.3. Spent Fuel Cooling and Cleanup Systems
>
> "The spent fuel cooling and cleanup systems for each unit are designed to maintain the water quality and clarity of the pool water and to remove the decay heat generated by the stored spent fuel assemblies. The cooling system is designed nonseismic and consists of redundant 100 percent capacity systems. The cooling water for the secondary side of the spent fuel pool cooling system heat exchangers is provided by the nonseismic station service water system. Regulatory Guides 1.13, "Fuel Storage Facility Design Basis," and 1.29, "Seismic Design Classification," guidelines state that the spent fuel pool cooling system and its secondary cooling be designed to seismic Category I requirements. The applicant has provided an analysis to show that the results of a failure of the cooling system do not result in a dose release exceeding a small fraction of 10 CFR Part 100 guideline limits. We performed an independent analysis that verified the applicant's results. Based on our analysis, we conclude that the alternative to a seismic Category I designed spent fuel pool cooling system is acceptable.
>
> "A permanently installed seismic Category I connection from the core standby cooling system-equipment cooling water system provides an alternate makeup water source to the pool."

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"Based on our review, we conclude that the design of the spent fuel pool cooling and cleanup system is in conformance with the requirements of Criteria 61 and 62 of the General Design Criteria, Branch Technical Position ASB 9-2 with respect to decay heat loads, and the positions in Regulatory Guides 1.13 and 1.29, including the positions on availability of assured makeup sources but excluding the position on seismic Category I classification which is justified by dose release analysis described above and is, therefore, acceptable."

Both of the LaSalle SFPs were re-racked for high density fuel storage in 1989 (Unit 2) and in 1993 (Unit 1). Since these re-racks provide the capability of storing more than 5 cores in the SFP, the NRC re-reviewed the spent fuel cooling capability, including the RHR availability. The Unit 1 rerack SER (Amendment 90 to Facility Operating License No. NPF-11, February 24, 1993) states in part:

"<u>2.2. Spent Fuel Pool Cooling System</u>

"The SFP cooling, filter, and demineralizer system for each unit contains two cooling pumps, two HXs and two filter/demineralizers (F/Ds). The two pumps are arranged in parallel, as are the HXs and F/Ds. The system is arranged so that flow from the pumps is directed first to the F/Ds and then to the HXs. When necessary, the F/Ds may be bypassed with the SFP coolant passing directly to the HXs and then returning to the pool."

***

"<u>2.2.5. Alternate Cooling Methods</u>

"<u>2.2.5.1. Use of Residual Heat Removal System</u>

"In the event the SFP cooling system is completely inoperable, the B train of the residual heat removal (RHR) system may be employed to cool the SFP once the full core has been unloaded into the pool.

"<u>2.2.5.2. Coolant Addition</u>

"In the event conditions occur which allow the SFP coolant to boil, normal makeup may be made from the cycled condensate storage system. This system, however, is not safety-related and may not be available, except during normal operating conditions. In the event that the cycled condensate storage system is not available, SFP coolant makeup may be provided by the core standby cooling water system-equipment water cooling system (CSCS-EWCS), which is intended as an emergency makeup water source for the SFP. The CSCS-EWCS is safety-related and meets the single-failure criterion."

***

"In view of the foregoing information, the staff finds that all SFP cooling concerns related to the proposed reracking have been adequately addressed."

The Unit 2 rerack SER (Amendment 48 to Facility Operating License No. NPF-18, June 15, 1989) states in part:

"2.2. Spent Fuel Pool Cooling System

"The spent fuel pool cooling system (SFPCS) consists of two identical trains of equipment. Each train consists of one 3000 gpm centrifugal pump and one 14.6 MBtu/Hr tube-and-shell heat exchanger. After water from the spent fuel pool is cooled by the heat exchangers, it is purified by the spent fuel pool cleanup system. Neither the SFPCS nor the cleanup system are seismic Category I. In the event of an excessive heat load, the "B" loop of the Residual Heat Removal (RHR) system can be used to cool the spent fuel pool. The RHR system, including all piping to and from the spent fuel pool, is independent of the SFPCS and is seismic Category I."

***

"2.2.2.1. Makeup Water

"...The SFPCS is not seismic Category I and it is not powered by a Class 1E source (i.e., on-site emergency diesel generator). Under such circumstances, SRP Section 9.1.3 identifies an alternative method for cooling of spent fuel following an earthquake.

"Specifically, the SRP discusses use of a seismic Category I spent fuel pool makeup water capability and a seismic Category I ventilation system to process potential radiological releases to the pool building resulting from pool boiling. The LaSalle FSAR identifies the emergency fuel pool makeup system (EFPMS) as the seismic Category I makeup water system for the spent fuel pool.

"The EFPMS includes two 300 gpm pumps and is part of the seismic Category I core standby cooling system-equipment cooling water system (CSCS-EWCS).

"<u>2.2.2.2. Building Ventilation</u>

"With regard to qualified ventilation capability when seismic Category I spent fuel pool cooling is not provided, the LaSalle FSAR identifies the standby gas treatment system (SGTS) as the qualified ventilation system. The SGTS is designed to seismic Category I criteria and consists of two redundant filter trains. This system is designed to remain operational during design basis events and is protected against natural phenomena.

"<u>2.2.4. Loss of Cooling</u>

"...The staff further concludes that the seismic Category I EFPMS and SGTS meet the requirements of GDC 2 for ensuring adequate spent fuel pool cooling and prevention of unacceptable radiological releases following an earthquake."

Based on the above, the LaSalle design and analyses have been, and are, consistent with the guidance described in Standard Review Plan, Section 9.1.3, "Spent Fuel Pool Cooling and Cleanup System."

<u>Response 12 Conclusion:</u>

The NRC has reviewed the design of the SFP and the FPCCS during their OL reviews and during the rerack reviews, and has determined that the SFP and the FPCCS meet the applicable licensing criteria. The 105 % power uprate does not affect the design of the SFP or of the FPCCS, except by a small increase in the amount of decay heat generated in the SFAs, and thereby a small increase in the decay heat load in the SFP which can be accommodated by the FPCCS. The major changes in the decay heat loads are a result of using bounding assumptions in all of the parameters involved in analyzing the "worst case" batch offload and emergency full core offload, as described in our response to NRC Question 8a. Even these bounding assumptions, which are deliberately chosen to provide conservative results, provide SFP peak temperatures that conform to the SRP 9.1.3 acceptance criteria.

Question 13:

In order to determine whether adequate controls exist to ensure the guidance of Standard Review Plan, Section, 9.1.3, are met, the staff needs to understand the provisions established or to be established in plant operating procedures to monitor and control the SFP water temperature during full-core offload events. Provide the following information:

a. How often the local temperature indicators for SFP water temperature will be monitored.

b. The setpoint of the high water temperature alarm for the SFP.

c. Information supporting a determination that there is sufficient time for operators to intervene in order to ensure that the temperature limit of 150°F will not be exceeded.

d. The mitigative actions (i.e. prohibit fuel handing, aligning other systems to provide SFP cooling, etc.) to be taken in the event of a high SFP water temperature alarm.

Question 13a:

How often the local temperature indicators for SFP water temperature will be monitored.

Response 13a:

During normal operation, including core offload refuelings, the SFP water temperature is monitored continuously (i.e., via process computer) and the process computer annunciates in the Control Room if the SFP temperature exceeds 100°F. The SFP temperature is recorded daily per surveillance procedure.

Question 13b:

The setpoint of the high water temperature alarm for the SFP.

Response 13b:

The SFP water temperature is monitored continuously (i.e., via process computer) and the process computer annunciates in the Control Room if the SFP temperature exceeds 100°F.

Question 13c:

Information supporting a determination that there is sufficient time for operators to intervene in order to ensure that the temperature limit of 150°F will not be exceeded.

Response 13c:

The LaSalle Technical Specifications define Operational Condition 5, Refueling, as average reactor coolant temperature ≤ 140°F. The SFP is normally maintained at less than 95°F to limit or minimize release of radioactive gases from the SFP water.

Pursuant to UFSAR sections 9.1.3.1.1 and 9.1.3.2.1, the FPCCS is designed to keep the SFP below 120°F during normal operation, with only one train of FPCCS in operation. During refueling modes, the SFP temperature is allowed to reach 150°F when the reactor cavity and dryer/separator pit are being drained and flow is being diverted, or when the decay heat load is approximately the UFSAR EHL value of 42×10^6 BTU/hr. The SFP water temperature is monitored continuously (i.e., via process computer) and the process computer annunciates in the Control Room if the SFP temperature exceeds 100°F. Therefore, the temperature "limit" of 150°F is viewed as an upper bound on operational evolutions, which would not normally be approached.

The bounding power uprate analyses, which as previously noted contain assumptions and parameters which were deliberately chosen to provide ultra-conservative decay heat loads, show that with both trains of FPCCS operating, even for an "emergency" full core offload commencing 24 hours after shutdown, the SFP temperature peaks at 150.5°F at 78 hours after shutdown. Since the SFP temperature would be annunciated by the process computer at 100°F, and would be trending upward for this scenario, there is time for operator intervention (e.g., provision of the other unit's FPCCS cooling capability, fuel movements into the SFP to be slowed down or halted, etc.). See Figure 8a-2 in our response to NRC Question 8a above.

The LaSalle FPCCS Abnormal Operating Procedure is invoked when one of the symptoms/entry conditions is a high SFP temperature (110°F), with the FPCCS in operation.

Question 13d:

The mitigative actions (i.e. prohibit fuel handing, aligning other systems to provide SFP cooling, etc.) to be taken in the event of a high SFP water temperature alarm.

Response 13d:

The SFP water temperature is monitored continuously (i.e., via process computer) and the process computer annunciates in the Control Room if the SFP temperature exceeds 100°F. The LaSalle FPCCS Abnormal Operating Procedure is invoked when one of the symptoms/entry conditions is a high SFP temperature (110°F). Fuel handling operations would be suspended if any abnormalities would occur during fuel movements (i.e., the procedure directs the operators to "place components in a safe condition"). The LaSalle FPCCS Abnormal Operating Procedure addresses a multitude of mitigative measures, including checking for leaks, restoration of power to components, provision of additional temperature and/or radiation monitoring, verification of SFP level, provision of additional SFP cooling by alternate means including aligning the other unit's FPCCS cooling capability; and emergency makeup from various sources.

Question 14:

The equipment qualification (EQ) of mechanical equipment with non-metallic components inside and outside containment has not been addressed. Please demonstrate that plant operations at the proposed uprated power level will have no impact on the EQ of mechanical equipment with non-metallic components inside and outside containment.

Response 14:

LaSalle's licensing commitment is to meet the requirements of 10CFR50.49, "Environmental Qualification of Electric Equipment Important to Safety for Nuclear Power Plants." Currently, only the electrical equipment within the defined scope of 10CFR50.49 is covered by the Station's Environmental Qualification (EQ) Program. The current EQ basis is not impacted by the power uprate and resulting environmental conditions. The EQ of active mechanical equipment containing non-metallic materials is ensured by sound design, periodic inspection, and testing, maintenance, and refurbishment when necessary. The adequacy of the organic materials used in the equipment is ensured through the procurement process (procurement specifications include environmental and process requirements) by choosing materials suitable for the applications and establishing replacement intervals based on vendor/industry experience.

The changes are negligible, for the normal and accident environmental conditions inside and outside containment, due to power uprate. In addition, the process temperatures and radiation effects from power uprate are within the pre-uprate design limits. The only process parameter of concern would be in the Feedwater System, whereby power uprate could increase the normal feedwater temperature from the present 420°F to 426.5°F, which is a negligible increase and within the original design temperature limit. Thus, the environmental qualification of the non-metallic materials in the mechanical equipment exposed to the power uprate process conditions is not adversely impacted.