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February 12, 1981

Mr. B. J. Youngblood, Chief
Licensing Branch 1
Division of Licensing
U.S. Nuclear Regulatory Commission
Washington, D. C. 20555

Subject: LaSalle County Station
Units 1 and 2 Resolution
of Reactor Systems Branch
Questions
NRC Docket Nos. 50-373/374
LOD 81-40-19

Dear Mr. Youngblood:

The purpose of this letter is to provide supplemental materials requested by the NRC Staff in order to resolve issues identified by the Reactor Systems Branch (RSB). Based on meetings with that Branch on February 5 and February 11, 1981 the questions requiring further action by the applicant have been reduced to the following:

1. Internally Generated Missiles - Enclosure 1
2. ODYN Reanalysis and MCPR Assessment - Enclosure 2
3. Control Rod Scram Discharge Volume System - Enclosure 3
4. Safety-Relief Valves - Enclosure 4
5. Post LOCA ECCS Leakage - Enclosure 5
6. LOCA Reanalysis - Enclosure 6
7. NUREG - 0737 Items - Enclosure 7
   a. II.D.1 S/RV Testing
   b. II.K.1(10) & (23) - I.E. Bulletin 79-08
   c. II.K.3 - B&O Task Force Recommendations
      (13) RCIC Auto Restart (Implementation Schedule)
      (21) HPCS Auto Restart
      (22) RCIC Suction (Implementation Schedule)
      (25) Emergency Power on Pump Seals
      (46) Michelson Concerns

All other issues raised by this Branch have been resolved based on material already documented by the applicant and are, therefore, not addressed in this letter.

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With respect to NUREG-0737 Items II.K.3(13) and II.K.3(22) there exists a need to request relief from the implementation schedules defined in the NUREG. The bases for the relief are defined in Sections L.2 and L.3 respectively of the FSAR and were reviewed with the NRC Staff management at the meeting of December 22, 1980. That justification is restated in the enclosure to this letter for purposes of completeness.

It has come to our attention that four issues previously under review by RSB now fall within the responsibility of the Auxiliary Systems Branch (ASB). These four issues are:

1. Internally Generated Missiles
2. Control Rod Scram Discharge Volume
3. Control Rod Return Line Modification
4. Inter-System Leakage

The first two items in this list are discussed in the enclosures and will not be addressed here. Item 3 had been reviewed and resolution reached on the basis of the commitment documented by the applicant in Section 4.6.1 of the FSAR to perform CRD seal leakage testing as part of the LaSalle County Unit 1 initial test program. Item 4 was ultimately resolved on the basis of periodic surveillance in the hi-point vents on all ECCS lines. This surveillance is documented in Section 4.5.1.a.1 of the Technical Specification. These latter two items are judged by the applicant to be resolved, and no further information is being prepared at this time for submittal to the NRC.

In the event you have any questions in this regard, please direct them to this office.

Very truly yours,

L. O. DelGeorge

L. O. DelGeorge
Nuclear Licensing
Administrator

Enclosures 1-7
cc: NRC Resident Inspector-LSCS

ENCLOSURE 1

INTERNALLY GENERATED MISSILES

The NRC Staff requested that an assessment be made to identify the limiting missiles with the potential for being generated in the containment. The potential for such missile generation has been discussed in Section 3.5.1.2 of the FSAR. It was concluded on the basis of the review discussed therein that no safety concern was presented by such potential missiles due to the segregation design of ESF systems. This segragation design was briefly reviewed by the NRC Reactor Systems Branch and has been discussed with the Auxiliary Systems Branch. The applicant is currently in the process of identifying, by a bounding missile survey, representative missile energies which will be compared to the allowable emergy absorption characteristics of the containment. The results of this assessment will be documented in Amendment 55 of the FSAR which will be submitted by March 1, 1981.

Enclosure 2

ODYN Reanalysis and MCPR Assessment

Extensive reanalysis of the LaSalle County pressurization transients has been done using the ODYN Code to resolve concerns expressed by the NRC Staff. The results of these analyses are provided as an attachment to this enclosure in the form of Table 15.0-2 and new Figure 15.0-3. The results reported that were calculated using ODYN are noted. The remainder of the transients, with the exception of the Rod Withdrawal Error event for which the code PANACEA[(a)] was used, were evaluated using the REDY code. These are all approved codes.

Furthermore, to resolve a question raised by the Staff relative to certain results characterized as estimates, all transient and accident shown in the Chapter 15 summary table were simulated by codes approved by the NRC for the specific type event. A second code used to calculate CPR's was not always employed. If the event was (1) obviously bounded by a more severe event of the same type [see Table 15.0-3 Note 2] or (2) there was obviously no threat to the MCPR safety limit due to the event being initiated at less than 100% power [see Table 15.0-3 Note 3]. Therefore, each of the CPR values reported represents an actual or bounding value for the event.

In addition, in response to a question from the Staff relative to the peak vessel pressure vs. S/RV capacity shown in Figure 5.2-3 the following information is provided:

1. The LaSalle County analysis was done for the all valve case.

2. Previous studies assure us that in every case (i.e. valve capacity for any number of valves including all valves), the ODYN result will be more conservative than the REDY result. Therefore, the result shown in Figure 5.2-3 is justifiably conservative for the 3/4 valve criterion allowed by the ASME code. In other words a 13 valve case for LaSalle County run by ODYN would remain significantly below the Code allowable because the REDY result for 13 valves is below the code allowable.

(a) PANACEA: J. A. Wooley, "3 Dimensional BWR Core Simulator" January, 1977. NEDO-20953A.

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TABLE 15.0-2

SUMMARY OF EVENTS RESULTS

Note: Paragraphs Denoted with Suffix A are Reanalyses WITH ODYN Code (See Reference 2 of 15.1.2.6).

PARAGRAPH	FIGURE	DESCRIPTION	MAXIMUM NEUTRON FLUX % NBR	MAXIMUM DOME PRESSURE psig	MAXIMUM VESSEL PRESSURE psig	MAXIMUM STEAM LINE PRESSURE psig	MAXIMUM CORE AVERAGE SURFACE HEAT FLUX % OF INITIAL	ΔCPR	FREQUENCY CATEGORY*	DURATION OF BLOWDOWN: NO. OF VALVES 1st BLOWDOWN	DURATION OF BLOWDOWN: DURATION OF BLOWDOWN sec.
15.1		DECREASE IN CORE COOLANT TEMPERATURE									
15.1.1	15.1.1-1	Loss of Feedwater Heater, AFC	111.4	1020	1058	994	106.1	0.06	a	0	0
15.1.1	15.1.1-2	Loss of Feedwater Heater, MFC	123	1030	1067	1017	117.2	0.16	a	0	0
Omit replace with 15.1.2A ODYN solution below					1170	1126	112.2	1.07	a		
15.1.2A	15.1.2-3	Feedwater Ctrl Failure, MAX DEMAND HI FOR W/BYPASS	215	1140	1163	1135	114.4	0.11	a	13	6
Omit 15.1.2A	~~15.1.2-4~~	~~Feedwater Ctrl Failure,~~	375	1167	1189	1166		0.11	b	13	6
15.1.3	15.1.3-1	Pressure Regulator Fail Open 115% Flow	103.9	1068	1083	1066	100.0	0.05	a	0	0
15.2		INCREASE IN REACTOR PRESSURE									
Omit replace with 15.2.2A ODYN Solution below Figure 15.2.2.3		~~Bypass-On, RPT-On~~	190	1140	1167	1124	106	1.12	a		
Omit replace with 15.2.2A ODYN Solution below Figure 15.2-2.4		tion Bypass-Off, RPT-On									
15.2-2A	15.2.2-3	Generator Load Rejection, Bypass-On, RPT-On	214	1140	1166	1131	106.4	0.07	a	18	
15.2.2A	15.2.2-4	Generator Load Rejection, Bypass-Off, RPT-On	350	1166	1192	1163	113.6	0.15	b	13	

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TABLE 15.0-2 (Cont'd)

SUBSEC-TION	FIGURE	DESCRIPTION	MAXIMUM NEUTRON FLUX % NBR	MAXIMUM DOME PRESSURE psig	MAXIMUM VESSEL PRESSURE psig	MAXIMUM STEAM LINE PRESSURE psig	MAXIMUM CORE AVERAGE SURFACE HEAT FLUX % OF INITIAL	ΔCPR ~~MINIMUM CPR~~	FREQUENCY CATEGORY*	DURATION OF BLOWDOWN: NO. OF VALVES 1st BLOW-DOWN	DURATION OF BLOWDOWN: DURA-TION OF BLOW-DOWN (sec)**
15.2.3	15.2.3-1	Turbine Trip, Bypass-On, RPT-On	163	1138	1164	1123	103.1	0.08 ~~[illegible]~~	a	15	5.4
Insert Replace with Offset Solution 15.2.3A below, Figure 15.2-3-3											
~~[illegible]~~	~~[illegible]~~	~~[illegible] RPT-On~~	~~[illegible]~~	~~[illegible]~~	~~[illegible]~~	~~[illegible]~~	~~[illegible]~~	~~[illegible]~~	~~[illegible]~~	~~[illegible]~~	~~[illegible]~~
15.2.3A	15.2.3-3	Turbine Trip, Bypass-Off, RPT-On	352	1163	1190	1162	112.1	0.13 ~~[illegible]~~	b	18	—
15.2.4	15.2.4-1	Main Steam Line Isolation, Position Switch	269	1163	1199	1152	108.6	<0.05(2) ~~[illegible]~~	a	18	5.4
15.2.5	15.2.5-1	Loss of Condenser Vacuum at 2 inches per sec	151	1134	1159	1120	104	<0.08(2) ~~[illegible]~~	a	14	6.[illegible]

TABLE 15.0-2 (Cont'd)

[illegible]	FIGURE	DESCRIPTION	MAXIMUM NEUTRON FLUX % NBR	MAXIMUM DOME PRESSURE psig	MAXIMUM VESSEL PRESSURE psig	MAXIMUM STEAM LINE PRESSURE psig	MAXIMUM CORE AVERAGE SURFACE HEAT FLUX % OF INITIAL	ΔCPR ~~MINIMUM CPR~~	FREQUENCY CATEGORY*	DURATION OF BLOWDOWN: NO. OF VALVES 1st BLOWDOWN	DURATION OF BLOWDOWN: DURATION OF BLOWDOWN sec.**
15.2.6	15.2.6-1	Loss of Auxiliary Power Transformer	103.9	1092	1103	1092	100.0	~ 0.0 ~~[illegible]**~~	a	2	5.6
15.2.6	15.2.6-2	Loss of All Grid Connections	150.4	1135	1161	1121	101.8	< 0.08 (2) ~~≥1.13**~~	a	16	5.4
15.2.7	15.2.7-1	Loss of All Feed-water Flow	103.9	1094	1105	1094	100.0	~ 0.0 ~~≥1.24**~~	a	2	5.5
15.3		DECREASE IN REACTOR COOLANT SYSTEM FLOWRATE									
15.3.1	15.3.1-1	Trip of One Recircu-lation Pump Motor	104.0	1020	1056	994	100.0	~ 0.0 ~~≥1.21**~~	a	0	0
15.3.1	15.3.1-2	Trip of Both Recircu-lation Pump Motors	103.9	1094	1107	1092	100.0	~ 0.0 ~~≥1.24**~~	a	2	5.3
15.3.2	15.3.2-1	Fast Closure of One Main Recirc Valve - [illegible]/sec	103.9	1095	1108	1093	100.0	~ 0.0 ~~≥1.24**~~	a	2	5.4
15.3.2	15.3.2-2	Fast Closure of Two Main Recirc Valves - 11%/sec	103.9	1095	1108	1099	100.0	~ 0.0 ~~[illegible]**~~	a	6	5.3
15.3.3	15.3.3-1	Seizure of One Recircu-lation Pump	103.9	1107	1119	1101	100.2	~ 0.0 ~~≥1.24**~~	c	6	5.0
15.4		REACTIVITY AND POWER DISTRIBUTION ANOMALIES									
15.4.4	15.4.4-1	Startup of Idle Recircu-lation Loop	100.2	982	995	971	79.2	< 0.18 (3) ~~≥1.06*~~	a	0	0
15.4.2	15.4.2-1	Rod withdrawal error at power	110.4	1020	1050	985	103.4	< 0.18	a	0	0

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TABLE 15.0-2 (Cont'd)

PARA-GRAPH	FIGURE	DESCRIPTION	MAXIMUM NEUTRON FLUX % NBR	MAXIMUM DOME PRESSURE psig	MAXIMUM VESSEL PRESSURE psig	MAXIMUM STEAM LINE PRESSURE psig	MAXIMUM CORE AVERAGE SURFACE HEAT FLUX % OF INITIAL	ΔCPR	FREQUENCY CATEGORY* (2)	DURATION OF BLOWDOWN: NO. OF VALVES 1st BLOWDOWN	DURATION OF BLOWDOWN: DURATION OF BLOWDOWN SEC.**
15.4.5	15.4.5-1	Fast Opening of One Main Bypass Valve- [illegible]/sec	281.8	980	1000	971	76.2	≤0.15	a	0	0
15.4.5	15.4.5-2	Fast Opening of Both Main Bypass Valves- [illegible]/sec	193.8	973	980	961	72.2	<0.15 (3)	a	0	0
15.5		INCREASE IN REACTOR COOLANT INVENTORY									
15.5.1	15.5.1-1	Inadvertent HPCS Pump Start	103.9	1020	1058	994	100.0	~0.0	a	0	0

* a = incidents of moderate freq; b = infrequent incidents; c = limiting faults.
** = estimated value.

(1) CPR Results without Adders

(2) Events Bounded By More Severe Transients, hence unique MCPR's are not calculated but a limiting value is used

(3) Events initiated From Low Power & Therefore resulting MCPR is well above the 1.06 MCPR Safety limit.

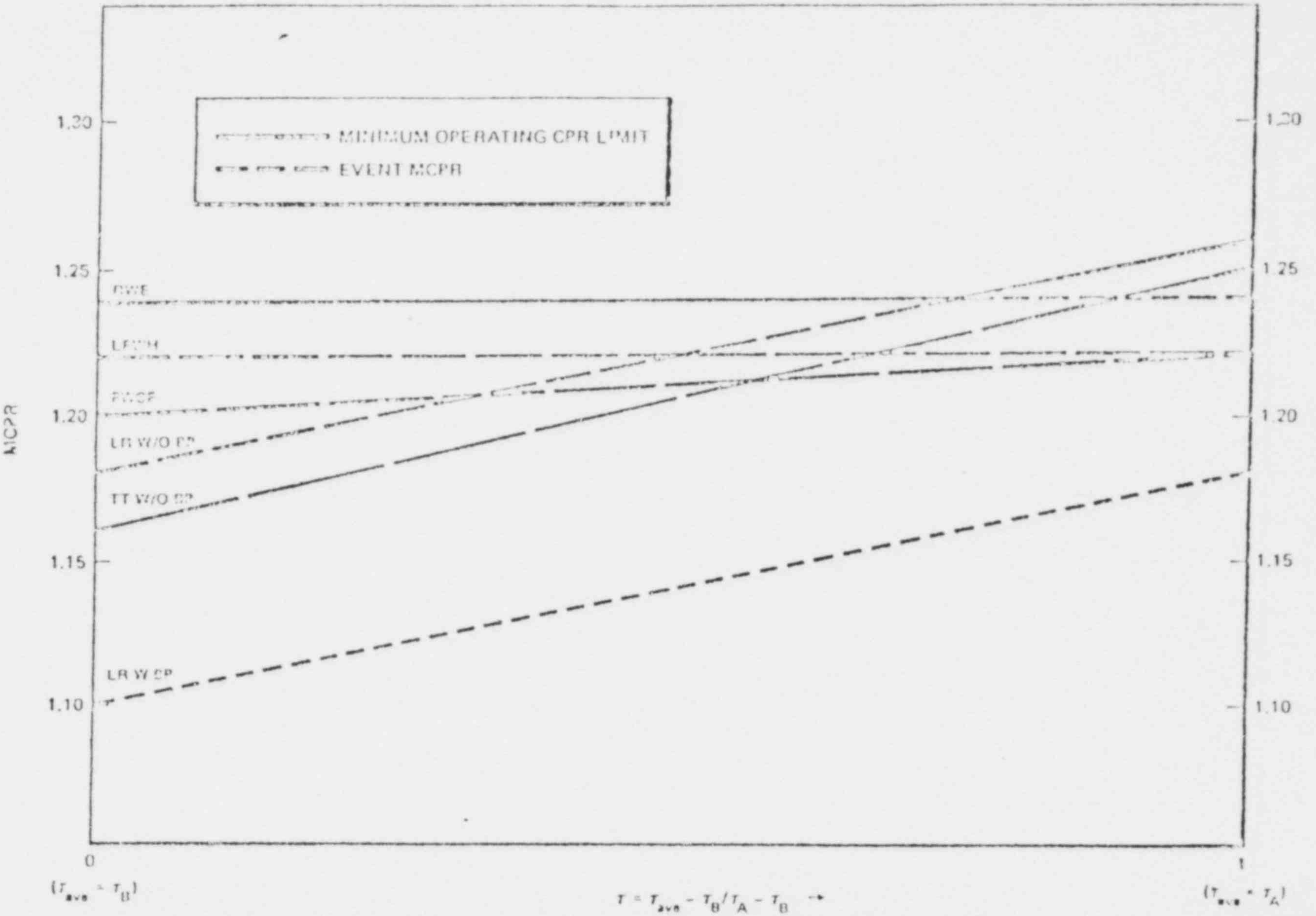

Figure 15.0-3 Minimum Operating CPR Limit

including the thermal limits given previously. The core and fuel design basis for steady-state operation, i.e., MCPR and LHGR limits, has been defined to provide margin between the steady-state operating condition and any ~~fuel damage condition to accommodate uncertainties and~~ to ensure that no fuel damage results, even during the worst anticipated transient conditions at any time in life. For LSCS, the steady-state MCPR operating limit is [illegible] ~~shown in Figure 15.0-1~~ ... in Table 15.0-1

4.4.1.4 Requirements for Transient Conditions

Transient Limits

The transient thermal limits are established such that no fuel damage is expected to occur during the most severe abnormal operating transient. Fuel damage is defined as perforation of the cladding that permits release of fission products (Section 4.2). Mechanisms that cause fuel damage in reactor transients are:

a. severe overheating of fuel cladding caused by inadequate cooling, and

b. fracture of the fuel cladding caused by relative expansion of the uranium dioxide pellet inside the fuel cladding.

For design purposes, the transient limit requirement is met if at least 99.9% of the fuel rods in the core do not experience boiling transition during any abnormal operating transient. No fuel damage is expected to occur even if a fuel rod actually experiences a boiling transition.

A value of 1% plastic strain of Zircaloy cladding is conservatively defined as the limit below which fuel damage from overstraining the fuel cladding is not expected to occur. Available data indicate that the threshold for damage is in excess of this value. The linear heat generation rate required to cause this amount of cladding strain is approximately 25 kW/ft in unirradiated fuel, but decreases with burnup to approximately 20.5 kW/ft at a local exposure of 40,000 MWd/Te.

4.4.1.5 Summary of Design Bases

situation, whether a transient or an accident.

In summary then, the steady-state operating limits have been established to ensure that the design basis is satisfied for the most severe abnormal operational ~~transient~~. There is no steady-state design overpower basis. An overpower which occurs during an abnormal operational transient must abide the plant ~~transient limits~~. Demonstration that the transient limits are not exceeded is sufficient to conclude that the design basis is satisfied.

MCPR Safety limit for the

CHAPTER 15.0 - ACCIDENT ANALYSES

15.0.1 Approach to Safety Analysis

This safety analysis evaluates the ability of the plant to operate within the regulatory guidelines without undue risk to the public health and safety. The analysis investigates the categories of events by type and expected frequency to delineate the limiting cases where the radiological consequences are significant. This approach ensures that a broad spectrum of initiating events is considered. It also enables the focusing of more detailed treatment of the radiologically important cases, while subordinating trivial and nondominant cases to lesser relative importance. A hypothetical ATWS event is also included at the request of the NRC. It has an extremely low probability of occurrence at LSCS.

In the treatment of specific safety cases initiated by typical plant events, the concept of expected frequency was mutually considered with the mechanisms of radiological release, to scope the safety risk associated with that particular event.

The initiating events were assigned one of the following expected frequencies based upon practical Commonwealth Edison Company (CECo) operating experiences with seven nuclear power stations: INSERT

A. TRANSIENTS

- Moderate frequency - Events which may occur during a calendar year to once per 20 years for a particular plant. Anticipated operational transients are in this frequency class.

- Infrequent incidents - Events which are expected to occur once during the lifetime of the plant (including those that may occur once every 20 to 100 years). Unexpected or abnormal operational transients are included in this frequency class.

B. ACCIDENTS

- Limiting fault frequency - refers to those incidents that are never expected to happen but for which safety analyses are arbitrarily made to represent upper bounds on the radiological consequences. The design-basis accident is included in this frequency class.

It should be noted, for example, that the frequency of an initiating event may be described by the term limiting fault frequency and not be characterized as the limiting fault per se.

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The safety analysis will evaluate events based on two categories: These two categories will be transients and accidents. Transients will be broken down into two subsets these being moderate and infrequent events. Moderate and infrequent events will be treated as one category with regards to establishing the Critical Power Ratio (CPR) Operating Limit.

The safety analysis presents two categories of events: transients and accidents. Transients are subdivided into two subsets: moderate frequency events and infrequent incidents. For the purposes of simplifying a summary presentation of results, however, all transients are tabulated and independently of frequency with regard to the Critical Power Ratio (CPR) Operating Limit. The accident results are also tabulated in the same summary, Table 15.0-2.

e. Reactor core coolant flow decrease could result in overheating of the cladding as the coolant becomes unable to adequately remove the heat generated in the fuel.

f. Excess of coolant investory could result in damage resulting from excessive moisture carryover to the main turbine.

g. Postulated radioactive release from a subsystem or component due to loss of integrity.

h. Postulated (anticipated) transients without scram which results from multisystem maloperation plus active component failures. This is a hypothetical situation with an extremely low probability of occurrence.

Those eight categories include all of the situations effects on the nuclear system caused by abnormal operational ~~transients~~ which might lead to degradation of the reactor fuel barrier or reactor coolant pressure boundary. The variation of any one of these parameters may affect another. For purposes of analysis in the FSAR, events are analyzed in groups, according to the initiating incident or event. For example, positive reactivity insertions resulting from sudden pressure increases are evaluated in the increase in reactor pressure classification.

The input parameters and initial conditions used for the transient and accident analyses are listed in Table 15.0-1.

15.0.3 Judgment ~~of~~ Nonacceptable Safety Results

For all transient, ~~moderate~~ and infrequent frequencies, ~~[illegible]~~ the following ~~are~~ is considered to be unacceptable safety results:

a. Release of radioactive material to environs in excess of 10 CFR 20 limits.

b. Reactor operation induced fuel clad failures.

c. Nuclear system stresses in excess of those allowed for in the transient classification by applicable industry codes.

d. Containment stresses in excess of those allowed for in the transient classification by applicable industry codes.

~~[illegible]~~

~~a. Release of radioactivity which results in dose consequences exceeding one-tenth of 10 CFR 100 values.~~

~~b. Fuel damage that precludes resumption of normal operation after a normal restart.~~

~~c. Generation of a condition that results in a consequential loss of function of the reactor coolant system.~~

~~d. Generation of a condition that results in a consequential loss of function of a necessary containment barrier.~~

For the design-basis accidents (limiting faults), the following are considered to be unacceptable safety results:

a. Release of radioactivity resulting in dose consequences in excess of 10 CFR 100 values.

b. Failure of fuel cladding sufficient to cause changes in core geometry such that core cooling would be inhibited.

c. Nuclear system stresses in excess of those allowed for the accident (faulted) classification by applicable industry codes.

d. Containment stresses in excess of those allowed for the accident classification by applicable industry codes when containment is required as a barrier.

e. Radiation exposure to plant operations personnel in the main control room in excess of 5 rem whole body, 30 rem inhalation dose and 75 rem skin dose.

15.0.4 Method of Analysis

Each transient or accident is discussed and evaluated in terms of a sequence of events from the initiating condition to final stabilized state. The normal operation of unfailed equipment and controls is assumed. Credit is taken for plant systems and reactor protection systems in their normal functioning mode. The operation of unfailed engineered safety features (ESF) is also included. The effect of a single operator error or a single failure of active equipment is also included in certain analyses; however, this is done on an <u>or</u> basis for transient evaluations. In the evaluation of these postulated events, the plant damage allowances or limits are the same as those for normal operation. The evaluation presented herein interprets the accidents and transients in consonance with the historical frequency classification for the initiating events, i.e., the ~~allowance~~ margin or

limit was permitted ~~imposed~~ on that initiating event frequency rather than on contingent or conditional frequencies of ~~the~~ multiple events invloved in certain sequences.

It is important to recognize that certain arbitrary accident scenarios require the application of single failures and operator errors. Others, such as ATWS, require multiple failures for the postulated end condition. In these ~~incidents~~ (accidents), the overall frequency ~~classification also needs modification~~ to a much lower probability. Credible frequency classes for these multifailure, multierror scenarios do not currently exist, hence ~~we need to~~ recourse to ~~use~~ the former more simple classification for the convenience of cataloging these very low probability transients.

Most events postulated for consideration are already the results of single equipment failures or single operator errors hypothesized during normal or planned plant operations. Typical operational equipment failures or operator errors that can initiate events important to safety are as follows:

a. Undesired opening or closing of a single valve (a check valve is not assumed to close against normal flow).

b. Undesired starting or stopping of a single component (a change of state is not assumed without an assignable cause).

c. Malfunction or maloperation of any single control device.

d. Single failure of any electrical component.

e. Single operator error event by one person.

In general, the analyzed events have specific numerical input parameters and initial (state) conditions as specified in Table 15.0-1. Analyses that assume data inputs different from these values are designated accordingly in their discussion and the specific parameters are defined therein for such cases.

Initial Power/Flow Operating Constraints for Analysis Purposes

The analytical basis for most of the transient safety analyses is the thermal power at rated core flow (100%) corresponding to (105% Nuclear Boiler Rated steam flow. This operating point is the apex of a bounded operating power/flow map which, in response to any ~~of the credible~~ abnormal operational transients, will yield the minimum pressure and thermal margins of any operating point within the bounded map. Referring to Figure 15.0-1, the apex of the bounded power/flow map is point A, the upper bound is the design flow control line (105% rod line A-D'), the lower bound is the zero power line H-J', the right bound is the rated valve

Historical these are Analysis values approximately

position line A-H', and the left bound is either the low pump speed, minimum valve position line D-J or the natural circulation line D'-J'.

The power/flow map, A-D'-J'-H'-A, represents the acceptable operational constraints for abnormal operational transient evaluations.

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Any other constraint which may truncate the bounded power/flow map must be observed, such as the recirculation valve and pump cavitation regions, the licensed power limit and other restrictions based on pressure and thermal margin criteria. For La Salle, the licensed power is 3323 Mwt (point A), the power/flow map is not truncated by the line B-C. Reactor operation must be confined within the boundary A-D'-J'-L-K-A. For a ~~restricted~~ operating power level, such as point F, to satisfy pressure margin criteria, the upper constraint on power/flow is correspondingly reduced to the rod line, such as line F-G', which intersects the power/flow coordinate of the new operating basis. In this case, the operating bounds would be F-G'-J'-J-L-K-F. Operation would not be allowed at any point along line F-M, left of point F, at the derated power but at reduced flow. If operation is restricted to point F by ~~the~~ MCPR limitation ~~operating limit~~, operation at point M (or right of it) would be allowed provided the MCPR limit is not violated.

derated

Consequently, the upper operating power/flow limit of the reactor is predicated on the operating ~~basis~~ constraint of the analysis and the corresponding constant rod pattern line.

Certain localized events are evaluated at other than the above mentioned conditions. Such conditions are discussed specifically for that appropriate event.

Core and System Performance

which is defined as the Safety Limit CPR for LaSalle!

Section 4.4 describes the various fuel failure mechanisms. An acceptable criterion was determined to be that 99.9% of the fuel rods in the core would not be expected to experience boiling transition (Reference 1). This criterion is met by demonstrating that TRANSIENTS (and accidents) do not result in a minimal critical power ratio (MCPR) less than 1.06. The reactor steady-state operating limit is derived by determining the decrease in MCPR for the most limiting event. All other events result in smaller MCPR decreases and are not reviewed in depth in this chapter. The [illegible] is calculated [illegible] transient core heat transfer analysis computer program. The computer program is based on a multinode, single channel thermal hydraulic model which requires simultaneous solution of the partial differential equations for the conservation of mass, energy, and momentum in the bundle, and which accounts for axial variation in power generation. The primary inputs to the model include a physical description of the

INSERT 15

INSERT B (p. 15.0-6)

The s..ady-state reactor operating limit is determined as follows:

1) The change in the critical power ratio (CPR) which would result in the safety limit CPR (1.06) being reached, is calculated for each event. These CPR values are shown in Table 15.0-2

2) The ΔCPR value for each event is then added to the Safety Limit CPR value (1.06) to yield the event-based MCPR, except for those events whose ΔCPR is calculated using ODYN.

3) For events whose ΔCPR is determined by OYDN (all rapid pressurization events), the event-based MCPR is determined in conjunction with NRC-additive correction factors, the ΔCPR, and the Safety Limit CPR. These correction factors are in Table 15.0-1.

These results are given in Figure 15.0-3 for limiting transients and accidents.

The operating limit MCPR is the maximum locus of values from this event MCPRs calculated with the above method. The maximum calculated MCPR is depicted by the solid line in Figure 15.0-3. Maintaining the CPR operating limit at or above this solid line assures that the LaSalle Safety Limit CPR of 1.06 is never violated.

rf/626
2/4/81

bund[illegible], and channel inlet flow, [illegible] enthal[illegible] pressure [illegible] generati[illegible] as functions of time[illegible] these parameters [illegible] by the reactor [illegible] analys[illegible] comp[illegible] progr[illegible] and are shown in the figures for [illegible] three [illegible] A detailed description of the [illegible] in Appendix [illegible] of Reference 2. The [illegible] condition [illegible] for [illegible] full power transient MCPR [illegible] is that the [illegible] bundle is operating at the MCPR limit 1.2[illegible] Maintaining MCPR greater than 1.06 [illegible] sufficient, but not necessary, condition to assure that no fuel damage occurs. This is discussed in Section 4.4.

For situations in which fuel damage is sustained, the extent of damage is determined by correlating fuel energy content, cladding temperature, fuel rod internal pressure, and cladding mechanical characteristics. The bases for these correlations are the fuel rod failure tests discussed in Section 4.4, and in Section 6.[illegible]

Barrier Performance

This section primarily evaluates the performance of the Reactor Coolant Pressure Boundary (RCPB) and the containment system during transients and accidents.

During transients that occur with no release of coolant to the containment only RCPB performance is considered. If release to the containment occurs as in the case of limiting faults, then challenges to the containment are evaluated as well.

Releases from containment are evaluated for those specific cases which involve source terms outside the primary containment boundary. The normal operation of the SGTS and the single point stack is covered in each case.

Reactor Coolant Pressure Boundary Damage

The only significant areas of interest for internal pressure damage are the high-pressure portions of the reactor coolant pressure boundary (the reactor vessel and the high-pressure pipelines attached to the reactor vessel). The overpressure below which no damage can occur is defined as the pressure increase over design pressure allowed by the applicable ASME Boiler and Pressure Vessel Code, Section III, Class 1, for the reactor vessel and the high-pressure nuclear system piping. Because the ASME Boiler and Pressure Vessel Code, Section III, Class 1, permits pressure transients up to 10% over design pressure, the design pressure portion of the reactor coolant pressure boundary meets the design requirement if peak nuclear system pressure remains below 1375 psig (110% x 1250 psig).

Peak fuel enthalpy (discussed in Subsection 4.3) is used to evaluate whether reactor coolant pressure boundary damage occurs as a result of reactivity accidents. If peak fuel enthalpy

~~Peak fuel enthalpy (discussed in Subsection 4.2) is used to evaluate whether reactor coolant pressure boundary damage occurs as a result of reactivity accidents. If peak fuel enthalpy~~ remains below 280 cal/g, no reactor coolant pressure boundary (RCPB) damage results from nuclear excursion accidents and therefore no other barriers are challenged.

Radiological Consequences

In this section, the consequences of radioactivity release during ~~the~~ three types of events: (1) ~~incidents of moderate frequency (planned operational transients)~~ transients, ~~(2) infrequent incidents (abnormal operational transients)~~, and (2) limiting faults or (design-basis accidents) are considered. For all events whose consequences are limiting, a detailed quantitative evaluation is presented. For non-limiting events, a qualitative evaluation is presented or results are referenced from a more limiting or enveloping case or event.

For limiting faults or (design-basis accidents) two quantitative analyses are considered:

a. The first is based on conservative assumptions considered to be acceptable to the NRC for the purposes of the worst case bounding event which determines the adequacy of the plant design to meet 10 CFR Part 100 guidelines. This analysis is referred to as the "design-basis analysis".

b. The second is based on realistic assumptions considered to reflect expected radiological consequences. This analysis is referred to as the "realistic analysis".

Results for both are shown to be within NRC guidelines.

Results

The results of analytical evaluations are provided for each event. In addition ~~critical parameters are~~ the result summary is shown in Table 15.0.2. From that table ~~a~~ comparison of the limiting event for ~~that~~ any particular category ~~of events can be made~~ a comparison can be made.

15.0.5 Meteorological Parameters

Atmospheric dilution factors (χ/Q's) are summarized here for use in this chapter. The atmospheric dilution factors for the conservative analyses are based on the diffusion models presented in U.S. NRC Regulatory Guide 1.3, Revision 2 (June 1974).

The atmospheric dilution factors at the 50th %ile for the realistic analyses have been derived from 2 years of onsite meteorological data.

TABLE 15.0-1

INPUT PARAMETERS AND INITIAL CONDITIONS FOR

TRANSIENTS AND ACCIDENTS

1.	Thermal power level, MWt	
	Analysis value	~~3400~~ 3559 (104.9% NBR)
2.	Steam flow, lb per hr	14.81×10^6 (104% NBR)
3.	Core flow, lb per hr	108.36×10^6
4.	Feedwater flow rate, lb per sec	4115
5.	Feedwater temperature, °F	420
6.	Vessel dome pressure, psig	1020
7.	Vessel core pressure, psig	1031
8.	Turbine bypass capacity, %NBR	25
9.	Core coolant inlet enthalpy, Btu per lb	529
10.	Turbine inlet pressure, psig	962
11.	Fuel lattice	8 x 8
12.	Core average gap conductance, Btu/sec-ft^2-°F	0.1662
13.	Core leakage flow, %	12
14.	Required MCPR operating limit	~~1.24~~ See Section 15.0.4 Figure 15.0.3
15.	MCPR Safety Limit	1.06
16.	Doppler coefficient (-)¢/°F	
	Nominal EOC-1	0.221
	Analysis data	0.221

35. ( ) CORRECTION FACTORS FOR RAPID PRESSURIZATION EVENTS (ODYN SOLUTIONS)

This table will show the adders for the ODYN solution of pressure transients to compensate for the inaccuracies involved in modeling.

1.2.6 References

1. R. Linford, "Analysis Methods of Plant Transient Evaluations for the General Electric Boiling Water Reactor," NEDO-10802, April 1973.

2. F. ODAR, "SAFETY EVALUATION FOR THE GENERAL ELECTRIC TOPICAL REPORT: QUALIFICATION OF THE ONE-DIMENSIONAL CORE TRANSIENT MODEL FOR BOILING WATER REACTORS, NEDO-24154 AND NEDE-24154-P, VOLUMES I, II AND III", 1980

Note: A similar reference to the ODYN SER will be included on each Reference list pertaining to events for which ODYN was used.

TABLE 15.0-2

SUMMARY OF EVENTS RESULTS

Note: Paragraphs Denoted with Suffix A are Reanalyses WITH ODYN Code (See Reference 2 of 15.1.2.6).

PARA-GRAPH	FIGURE	DESCRIPTION	MAXIMUM NEUTRON FLUX % NBR	MAXIMUM DOME PRESSURE psig	MAXIMUM VESSEL PRESSURE psig	MAXIMUM STEAM LINE PRESSURE psig	MAXIMUM CORE AVERAGE SURFACE HEAT FLUX % OF INITIAL	ΔCPR ~~MCPR~~	FREQUENCY CATEGORY*	DURATION OF BLOWDOWN: NO. OF VALVES 1st BLOWDOWN	DURATION OF BLOWDOWN: DURATION OF BLOWDOWN SEC.
15.1		DECREASE IN CORE COOLANT TEMPERATURE									
15.1.1	15.1.1-1	Loss of Feedwater Heater, AFC	111.4	1020	1058	994	106.1	0.06 ~~1.17~~	a	0	0
15.1.1	15.1.1-2	Loss of Feedwater Heater, MFC	123	1030	1067	1017	117.2	0.16 ~~1.07~~	a	0	0
Omit, replace with 15.1.2A ODYN solution below											
~~15.1.2~~	~~15.1.2-[illegible]~~	~~Feedwater Cntl Failure, MAX DEMAND HI PWR~~	~~[illegible]~~	~~[illegible]~~	~~1170~~	~~1126~~	~~112.2~~	~~1.07~~	~~a~~	~~[illegible]~~	~~[illegible]~~
15.1.2A	15.1.2-3	Feedwater Cntl Failure, MAX DEMAND HI PWR W/Bypass	215	1140	1168	1135	114.4 ~~11[illegible]~~	0.11 ~~1.[illegible]~~	a	13	6
Omit 15.1.2A	~~15.1.2-4~~	~~Feedwater Cntl Failure,~~	~~375~~	~~1167~~	~~1189~~	~~1166~~	~~118.9~~	0.11 ~~1.[illegible]~~	~~b~~	~~13~~	6
15.1.3	15.1.3-1	Pressure Regulator Fail Open 115% Flow	103.9	1068	1083	1066	100.0	0.05 ~~1.[illegible]~~	a	0	0
15.2		INCREASE IN REACTOR PRESSURE									
Omit replace with 15.2.2A ODYN Solution below Figure 15.2.2.3		~~Generator Load Rejection, Bypass-On, RPT-On~~	~~155~~	~~1110~~	~~1162~~	~~1124~~	~~106~~	~~1.12~~	~~a~~	~~[illegible]~~	~~[illegible]~~
Omit replace with 15.2.2A ODYN Solution below Figure 15.2.2.4		~~Generator Load Rejection Bypass-Off, RPT-On~~	~~[illegible]~~	~~[illegible]~~	~~[illegible]~~	~~[illegible]~~	~~113.0~~	~~1.07~~	~~b~~	~~[illegible]~~	~~[illegible]~~
15.2.2A	15.2.2-3	Generator Load Rejection, Bypass-On, RPT-On	214	1140	1166	1131	106.4	0.07 ~~1.[illegible]~~	a	19	
15.2.2A	15.2.2-4	Generator Load Rejection, Bypass-Off, RPT-On	350	1166	1192	1163	113.6	0.15 ~~1.[illegible]~~	b	13	

TABLE 15.0-2 (Cont'd)

PARA-GRAPH	FIGURE	DESCRIPTION	MAXIMUM NEUTRON FLUX % NBR	MAXIMUM DOME PRESSURE psig	MAXIMUM VESSEL PRESSURE psig	MAXIMUM STEAM LINE PRESSURE psig	MAXIMUM CORE AVERAGE SURFACE HEAT FLUX % OF INITIAL	ΔCPR	FREQUENCY CATEGORY	DURATION OF BLOWDOWN: NO. OF VALVES 1st BLOW-DOWN	DURATION OF BLOWDOWN: DURA-TION OF BLOW-DOWN SEC**
15.2.3	15.2.3-1	Turbine Trip, Bypass-On, RPT-On	163	1136	1164	1123	103.1	0.08	a	12	5.4
[illegible] Selection 15.2.3A before Figure 15.2.3-3											
15.2.3A**	15.2.3-3	Turbine Trip, Bypass-Off, RPT-On	352	1165	1190	1162	112.1	0.13	b	13	—
								<0.03[2]			
		[illegible] at 2 [illegible]	[illegible]	1136	1159	1120	104	<0.03[2]	a	[illegible]	[illegible]

TABLE 15.0-2 (Cont'd)

PARAGRAPH	FIGURE	DESCRIPTION	MAXIMUM NEUTRON FLUX % NBR	MAXIMUM DOME PRESSURE psig	MAXIMUM VESSEL PRESSURE psig	MAXIMUM STEAM LINE PRESSURE psig	MAXIMUM CORE AVERAGE SURFACE HEAT FLUX % OF INITIAL	ΔCPR	FREQUENCY CATEGORY*	DURATION OF BLOWDOWN: NO. OF VALVES 1st BLOWDOWN	DURATION OF BLOWDOWN sec.**
15.2.6	15.2.6-1	Loss of Auxiliary Power Transformer	103.9	1092	1103	1092	100.0	~0.0	a	2	5.6
15.2.6	15.2.6-2	Loss of All Grid Connections	130.4	1135	1161	1121	101.8	<0.03 (2)	a	15	0.4
15.2.7	15.2.7-1	Loss of All Feedwater Flow	103.9	1094	1105	1094	100.0	~0.0	a	2	5.5
15.3		DECREASE IN REACTOR COOLANT SYSTEM FLOWRATE									
15.3.1	15.3.1-1	Trip of One Recirculation Pump Motor	104.0	1020	1058	994	100.0	~0.0	a	0	0
15.3.1	15.3.1-2	Trip of Both Recirculation Pump Motors	103.9	1094	1107	1092	100.0	~0.0	a	2	5.3
15.3.2	15.3.2-1	Fast Closure of One Main Recirc Valve - %/sec	103.9	1095	1108	1093	100.0	~0.0	a	2	5.4
15.3.2	15.3.2-2	Fast Closure of Two Main Recirc Valves - 11%/sec	103.9	1095	1108	1099	100.0	~0.0	a	6	5.3
15.3.3	15.3.3-1	Seizure of One Recirculation Pump	103.9	1107	1119	1101	100.2	~0.0	c	6	5.6
15.4		REACTIVITY AND POWER DISTRIBUTION ANOMALIES									
15.4.4	15.4.4-1	Startup of Idle Recirculation Loop	100.2	982	995	971	79.2	<0.18 (3)	a	0	0
15.4.2	15.B-1	Rod withdrawal error at power	110.4	1020	1050	985	103.4	<0.15	a	0	0

TABLE 15.0-2 (Cont'd)

PARA-GRAPH	FIGURE	DESCRIPTION	MAXIMUM NEUTRON FLUX % NBR	MAXIMUM DOME PRESSURE psig	MAXIMUM VESSEL PRESSURE psig	MAXIMUM STEAM LINE PRESSURE psig	MAXIMUM CORE AVERAGE SURFACE HEAT FLUX % OF INITIAL	ΔCPR	(3) FREQUENCY CATEGORY*	DURATION OF BLOWDOWN: NO. OF VALVES 1st BLOW-DOWN	DURATION OF BLOWDOWN: DURA-TION OF BLOW-DOWN SEC.**
15.4.5	15.4.5-1	Fast Opening of One Main Bypass Valve-20%/sec	231.8	980	1000	971	76.2	< 0.13	a	0	0
15.4.5	15.4.5-2	Fast Opening of Both Main Bypass Valves-11%/sec	193.8	973	980	961	72.0	< 0.18 (3)	a	0	0
15.5		INCREASE IN REACTOR COOLANT INVENTORY									
15.5.1	15.5.1-1	Inadvertent HPCS Pump Start	103.9	1020	1058	994	100.0	~ 0.0	a	0	0

* a = incidents of moderate freq; b = infrequent incidents; c = limiting faults.
** = estimated value.

(1) ODYN Results without ADDERS

(2) Events bounded by more severe transients, hence unique MCPR's are not calculated but a limiting value is used

(3) Events initiated from low power & therefore resulting MCPR is well above the 1.06 MCPR Safety limit.

ENCLOSURE 3

CONTROL ROD SCRAM DISCHARGE VOLUME SYSTEM

The LaSalle County Scram Discharge Volume (SDV) System has been reviewed in depth to ascertain the extent to which system modification is required to satisfy the design criteria developed by the BWR Owners Group and accepted by the NRC Staff. The LaSalle County design represents a marked improvement over that installed at Browns Ferry and is, therefore, judged by the applicant to justify the operation of LaSalle County Unit 1. The SDV system as it will exist at the time of Unit 1 fuel loading is shown in Figure E-1, and will hereafter be referred to as the "current" design.

The current design consists of two separate discharge volume headers, the piping for which is 10" dia. There exists an intergral instrumented volume 12" dia" at the end of these 10" headers. The instrument volume now provides redundant high level alarm and scram instrumentation. The scram instruments being of the Magnetrol design. The drain from the two instrumented volumes, which is a 2" dia pipe, is common to the two volumes, as is the single vent line. The vent line is vented to atmosphere. Both the vent and drain lines currently have a single valve to isolate the SDV.

Modifications to the system are planned for completion upon receipt of qualified equipment to provide (1) diverse level instrumentation (ΔP Transmitters) on each of the instrumented volumes, and (2) a second valve on the system vent line. The taps for the additional level instruments will be completed prior to fuel loading to facilitate the completion of the modificaiton later.

Because of the size of the header system, the fact that slopes are provided to assure proper drainage, that each header has an integral instrumented volume with redundant instrumentation, and the vent, which vents to atmosphere, is routed separately from the drain results in the applicants conclusion that the system is adequated to justify initial operation on LaSalle County Unit 1. Furthermore, because of the size and location of the instrumented volume, UT monitoring of header level is judged to be unnecessary.

This discussion with details concerning component design and performance characteristics will be documented in Amendment 55 of the FSAR to be submitted by March 1, 1981.

THE PRE-FUEL LOAD CONFIGURATION PROVIDES SAFETY EQUIPMENT REDUNDANCY FOR THE SCRAM DISCHARGE VOLUME HIGH LEVEL SCRAM. IT ALSO PROVIDES SEPARATION OF THE VENT AND DRAIN PIPING TO OVERCOME THE BROWN'S FERRY DEFICIENCY. THE BASIC DESIGN OF LASALLE PROVIDED COMPLIANT FEATURES AS FOLLOWS AS RECOMMENDED BY THE OWNER'S GROUP CRITERIA:

1) ADEQUATELY SIZED SCRAM DISCHARGE VOLUMES, A FULL SCRAM CAPABILITY EXISTS AFTER SDV HIGH LEVEL IS ATTAINED.
2) SEPARATION OF SCRAM DISCHARGE VOLUMES FOR EACH BANK OF HYDRAULIC CONTROL UNITS
3) HIGH POINT VENTING THROUGH AN ATMOSPHERIC GAP WITH VENTS AND DRAINS SEPARATED COMPLETELY SO NO INTERACTION IS POSSIBLE.
4) INTEGRAL INSTRUMENT VOLUMES WITH NO REDUCTION IN PIPE CROSS-SECTION BETWEEN SCRAM DISCHARGE HEADERS AND THE INSTRUMENT VOLUMES.
5) PIPING LAYOUTS TO GIVE GRAVITY DRAINAGE WITHOUT FLUID TRAPPING THROUGHOUT ENTIRE CRD PIPING LAYOUT.
6) SEPARATION OF SDV PIPING FROM CRD DRIVE & DRAIN PIPING ASSOCIATED WITH NORMAL CRD FUNCTIONS

A DIAGRAMMATIC REPRESENTATION OF THE LASALLE CRD SYSTEM IS ATTACHED TO SHOW THE PRE FUEL LOAD CONFIGURATION AND THE SUBSEQUENT CONFIGURATION WHICH ADDS THE FOLLOWING CHARACTERISTICS:

1. DIVERSE INSTRUMENT VOLUME SENSORS, WHEREAS THE PRE-FUEL LOAD CONFIGURATION HAD REDUNDANT FLOAT-TYPE LEVEL SWITCHES, THE subsequent configuration HAS FLOAT-TYPE AND

13 FEBRUARY 1981
FIELD SKETCHES

LASALLE COUNTY STATION UNITS 1 & 2

MODIFICATIONS TO CRD & SCRAM DISCHARGE VOLUME INSTRUMENTS

BACKGROUND

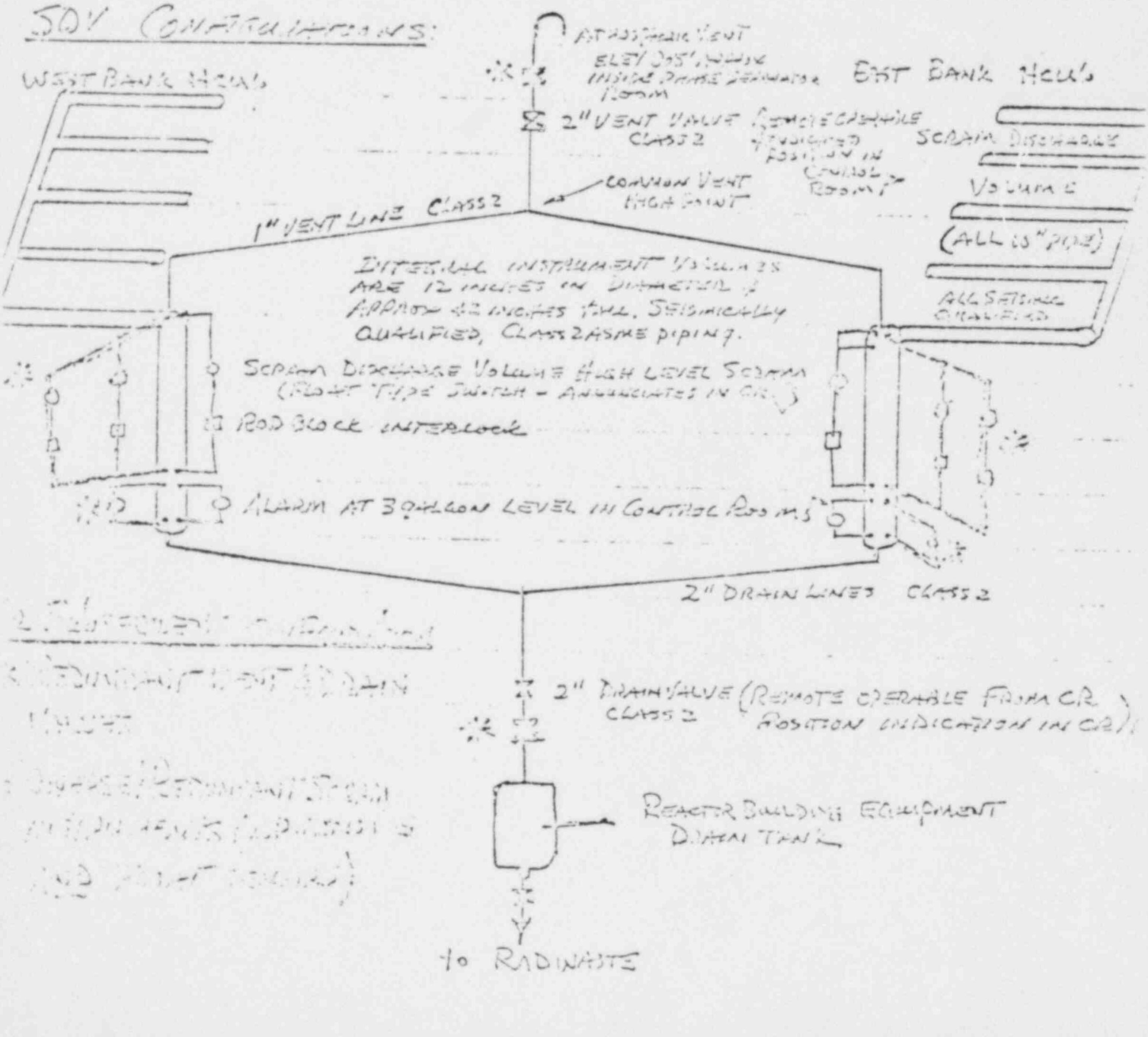
AS A RESULT OF THE BROWNS FERRY 3 INCIDENT INVOLVING FAILURE OF ALL RODS TO FULLY INSERT ON A SCRAM, THE LASALLE CRD SYSTEM WAS REVIEWED BY EDISON AND THE GENERAL ELECTRIC COMPANY'S LEAD SYSTEM ENGINEER. DISCUSSIONS WERE HELD WITH REGION III CONCERNING ALL EDISON BWR PLANTS AND EDISON PARTICIPATED WITH THE BWR OWNER'S GROUP IN DEFINING CRITERIA WHOSE FULFILLMENT WOULD PRECLUDE THE BROWNS FERRY 3 TYPE OF PROBLEM. A RESPONSE WAS MADE TO THE NRC'S BULLETIN 80-17 WHICH INDICATED THAT ALL EDISON BWR'S WOULD INSTALL MODIFICATIONS TO CONFORM TO THESE CRITERIA FROM THE UTILITY SPONSORED OWNERS GROUP. THE LASALLE OL IS PENDING THRUOUT THIS TIME PERIOD HENCE THE NEED TO DOCUMENT THE UNIQUE MODICATIONS FOR LASALLE UNITS 1 & 2 IN THE FSAR. THIS FIELD MEMO IS WRITTEN TO PROVIDE THE NRC STAFF REVIEWER WITH THE INFORMATION WHICH IS BEING EDITED FOR INCLUSION IN THE NEXT AMENDMENT TO THE FSAR (No. 55).

ATTENTION IS INVITED TO THE FACT THAT THE LASALLE MODIFICATIONS ARE TIME-PHASED INTO TWO PERIODS TO ENABLE A PRE-FUEL LOAD MODIFICATION THAT CAN BE MADE WITH AVAILABE HARDWARE AND A SEQUENTIAL POST-FUEL LOAD MODIFICATION THAT MUST AWAIT THE AVAILABILITY OF CLASS IE QUALIFIED CONTROL EQUIPMENT AND ASME CLASS 2 QUALIFIED VALVES. THE LEAD TIME FOR THESE QUALIFIED ITEMS DICTS THEIR INSTALLATION FOR THE FIRS

IS PROVIDED IN THE CONTROL ROOM.

2. REDUNDANT VENT AND DRAIN VALVES POWERED FROM SEPARATE DIVISIONS OF ESSENTIAL (SAFETY GRADE) POWER. THIS PROVIDES THE ISOLATION OF THE SCRAM DISCHARGE VOLUMES DURING THE ACTIVE SCRAM FUNCTION.

SDV CONFIGURATIONS:



ENCLOSURE 4

SAFETY-RELIEF VALVES

A question arose relative to the adequacy of the safety-relief valves at LaSalle County Station for operation during the "alternate shutdown" scenario. This issue was discussed in some depth in question Q212.146. The NRC Staff requested that supplemental information be provided relative to the manner in which valve flow rates were determined. This information was provided informally on December 17, 1980 and a copy is attached here for completeness. The attached information will be documented in a future amendment to the FSAR.

QUESTION 212.146

"Section 15.2.9 of the La Salle FSAR considered alternate shutdown cooling methods in the event the residual heat removal (RHR) system in the suction line may not be used because of valve failure. In the analysis, valves in the automatic depressurization system (ADS) were used to transfer fluid (steam, water or a combination of these) from the reactor vessel to the suppression pool. The RHR system removes the added heat by cooling water removed from the suppression pool and injecting it into the reactor vessel.

We require that you perform a test or cite previous test results to demonstrate that the ADS valves can discharge the fluid flow under the most limiting conditions when the fluid is all water. Show that this alternate method is a viable means of shutdown cooling by comparing the system hydraulic losses with the available pump head. Hydraulic losses should be provided for each system component and, wherever possible, should be derived from experimental results."

RESPONSE

Elevation head of water in the reactor vessel was taken to be 5 feet above the steamline nozzles. Suppression pool water is pumped into the vessel by a low-pressure pump (RHR-C) until the steamlines are flooded. The resultant enthalpy per pound of homogeneous vessel water is 204 Btu/lb. Saturation temperature of this water is 235° F, and the saturation (dome) pressure is 23.26 psia. The elevation of the relief valve outlet nozzles is about 24 feet below the steamline discharge nozzles, providing a small amount of subcooling at the relief valve inlet nozzle.

Calculation of flow per valve assumes no liquid back-pressure from the relief valve discharge line as the line-flow area is much larger than the port size. Some minor flashing liquid will cause a reduction in the expected liquid flow rate. The reduced flow rate for this slightly subcooled liquid is handled as such by a correction factor obtained from Fisher Control Handbook page 81, first edition, which provides a calculational method for such a fluid state.

The flow rate per valve is calculated to be about 1400 gpm under the conditions stated.

The suppression pool cooling mode of the RHR pump has a total flow resistance head of approximately 330 feet including line losses, static head, heat exchanger losses, inlet and

ATTACHMENT-3L

outlet losses at the pump and stainers. At this head, the pump capacity is over 7000 gpm for one RHR pump. The ADS valve flow rate when fed from the main steamlines is 1400 gpm per valve. Logically, then five open ADS valves given equivalent coolant flow to the regular LPCI system.

Also see the response to Question 212.132 which shows that a minimum of two ADS valves will always be available. The coolant flow for that minimal case is therefore 2800 gpm.

INSERT ATTACHMENT-3L INTO HERE.

Alternate Shutdown

Determine Flow Per Valve

A. Evaluate using valve data and equation below.

$$C_v = \frac{W}{3\ \frac{P_1}{2}} \quad \text{When } \Delta P \geq 1/2 \text{ P upstream}$$

$$C_v = \frac{1.05 \times 10^6}{3 \times \frac{1255}{2}} = 560.7 \rightarrow \underline{\text{use } 500}$$

1.05×10^6 → ASME Steam Flow for S/RV

1255 → Steam Design Inlet Press

Dome press Psia	Valve Inlet press Psia	Outlet press Psia	ΔP	Sat. Temp °F	Act Temp °F	ΔTemp	K_1	ΔP PSI K_1 (inlet)
15	30	20	10	250	219	31	.30	9
105	120	20	100	341	219	122	.675	81
85	100	20	80	328	219	109	.64	64
65	80	20	60	312	219	93	.60	48
45	60	20	40	293	219	74	.53	31.8
25	40	20	20	267	219	48	.41	16.4

$$Q = C_v \sqrt{\frac{\Delta P}{Sp.Gr.}} = \frac{500\sqrt{\Delta P}}{\sqrt{Sp.Gr.}}$$

Valve Indicated ΔP	Avail ΔP PSI	Sp. Gr.	Q/Value GPM	#From each valve	2 valves	3 valves
10	9	.958	1534	12229	24458	36687
100	81	.958	4602	36087	73374	110061
80	64	.958	4091	32613	65227	97839
60	48	.958	3543	28245	56489	84735
40	31.8	.958	2884	22991	45982	68973

Crossplot this on pump input plot. SEE ATTACHED PUMP INPUT PLOT

January 20, 1981

B. Reading Crossplot

	1 Value	2 Value	3 value
Dome pressure psia	122	[illegible]	31
Valve flow, #/min	39500	52000	56000
Pump flow in gpm	4795	6313	6793

C. Check to see if we can keep dome from filling up.

Elevation of steam lines = 53'9"
Elevation of Relief valve outle = 16.5" + 10" + 13" + 27'9"
$\sim$ 30'

Δ ELEVATION = 53.75' − 30' = 23.75 ft.

If C_v = 550/valve & water is @ 200°F, evaluate flow at 23.75' elevation head and 0 back pressure (psig).

Reevaluate vessel water temperature figuring 1/2 vol to fill to steam lines

Decay heat $\sim 9.5 \times 10^6$ BTU added

$$\text{Vesel enthalpy} = \left(\frac{11095\ ft^3}{.01701\ ft^3}\right) 218.8$$

$$= 142.7 \times 10^6 \text{ BTU}$$

$$\text{New added fluid} = 94\left(\frac{3750}{.01623}\right) = 21.7 \times 10^6 \text{ BTU}$$

$$\text{New total enthalpy} = 142.7 \times 10^6 + 21.7 \times 10^6 + 9.5 \times 10^6$$

$$= 173.9 \times 10^6$$

$$\text{New Mass} = \frac{11095}{.01701} + \frac{3750}{.01623} = .88 \times 10^{6}\#$$

Enthalpy/# = 179.3 x 106/.88 X 10^6 = 203.7 BTU/# =
SAT temp = 236°F SAT press 23.26 psia dome press.

$\frac{24 \text{ ft elevation}}{.01689 \times 144}$ head $\simeq$ 9.87 psi + vessel dome press = 33.13 psia

SAT temp at 33.13 psia = 256°F
ACT temp in vessel 236°F
20°F

From Fisher book p 81, K_c = .225

.225 (33.13) = 7.5 psi

$$Q = 550\sqrt{\frac{\Delta P}{S.G.}} \qquad S.G = .950$$

$$Q = 550\sqrt{\frac{7.50}{.948}} \text{ or 1547 gpm/value}$$

C. (Cont'd.)

C_v		550	Fill to 5' over steam lines
1 valve	=	1547 gpm	1610
2 valves	=	3093 gpm	3220
3 valves	=	4540 gpm	4830
4 valves	=	6187 gpm	6440
5 valves	=	7734 gpm	8050

Sensible heat /1°F

At Mode D-1 - 238 plant

= residual heat sensible heat @ 100°F/hr + RHR pump heat = (143.1×10^6) 2 loops

- remove residual heat and calc remainder

1 RHR pump @ 1200 BHP $\times 42.44 \frac{BTU}{min} \times 60 \frac{min}{hr}$

$$Q_p = 3.05 \times 10^6 \text{ BTU/hr - pump}$$

$$\text{Residual heat} = 3579 \times 10^6 \text{ Watts} \times .011 \times 3.419 \frac{BTU}{hr\ watt}$$

$$\text{Resid ht} = 134.3 \times 10^6 \text{ BTU/hr}$$

$$\text{Total heat} = 143.1 \times 10^6 \times 2 = 286.2 \times 10^6 \text{ BTU/hr}$$

$$\text{Sensible heat} = 286.2 \times 10^6 - 134.3 \times 10^6 - 2\,(3.05 \times 10^6)$$

$$\text{Sens.} = \frac{145.8 \times 10^6 \text{ BTU}}{100°} = 1.458 \times 10^6 \frac{BTU}{°F}$$

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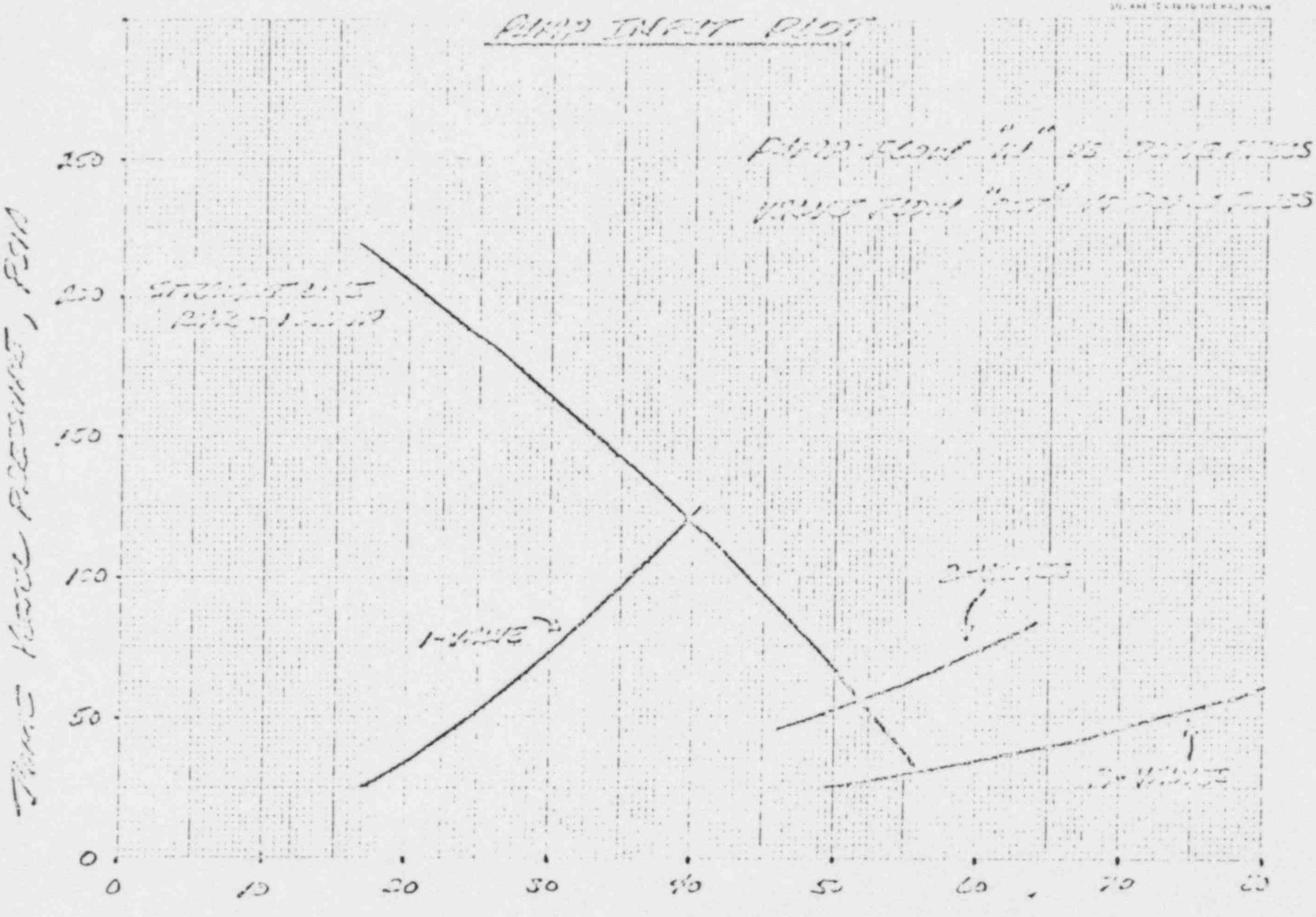
1/29/81

ALTERNATE SHUTDOWN

PUMP INPUT PLOT



ENCLOSURE 5

POST-LOCA ECCS LEAKAGE

Review procedure III.20 of SRP 6.3 requires that long term cooling capability following a LOCA should be adequate in the event of failure of any single active or passive component of the ECCS. The NRC Staff required the applicant to discuss how leakage from the first isolation valve in an ECCS suction line from the suppression pool during post LOCA long term cooling will be contained. The expressed concern is drainage of the suppression pool (heat sink) in view of the possible inaccessability for repair of leaking valve due to local contamination.

The scenaria postulated has been reviewed and is judged not to present a safety concern in light of the LaSalle County design. All of the ECCS suction valves are designed with a back seat to allow valve repacking at full system pressure with the valve open. In addition, the limit switches and torque switches have been adjusted at the factory to insure that in closing and opening, the travel is stopped by the torque switch with the limit switch set to shut off for backup safety. This means that anytime the ECCS suction valves are in the full open position (normal position) that the packing glands are isolated from the suppression pool water by the valve back seats. For these reasons, leakage of the type postulated is not expected to occur. In addition, the leakage monitoring program defined in Section L.37 of the FSAR to address NUREG-0737 Item III.D.1.1 will maintain leakage rates for these valves to an "as low as practicable" level. Alternate sources of water (condensate storage tank, service water. etc,) are available to assure adequate heat sink capacity by refilling the pool should excessive leakage occur. This information will be documented as an addendum to the response provided to Q212.37 of the FSAR.

ENCLOSURE 6

LOCA REANALYSIS

The revision to NRC Question 212.143 submitted with Amendment 50, October 1980, provided the PCT impact for a realistic conservative FCV closure rate of 11% second following LOCA. As previoulsy stated, the probability of any closure is exteremely unlikely. However, to respond to NRC concerns, the impact of an 11% second closure was provided. It should be noted that two failures would be required to initiate this closure. They are a failure of drywell high pressure signal, which locks the valve in position following a major LOCA, and a failure of the electronic valve control.

Recent telephone conversations with the NRC have led to the belief that an 11% second closure rate would be considered realistically conservative if the pressure sensors were properly qualified for a LOCA environment. It has been determined that these drywell pressure sensors are located on the instrument racks mounted outside the drywell and are connected to instrument lines originating in the drywell. The sensors are also indentical in design and manufacture to the pressure transmitters which provide a high drywell pressure signal to the RPS and ECCS systems.

The proposed revision to Q212.143 which documents this conclusion is provided as an attachment and will be formally submitted in a future amendment to the FSAR.

Enclosure 7

NUREG-0737 Items

a. II.D.1 S/RV Testing

Commonwealth Edison is a participant in the BWR Safety/Relief Valve Testing Program now under active review by the NRC Staff. The formal commitment to participation in and adherence to the results of this test program are documented in the L. O. DelGeorge letter to B. J. Youngblood (LOD 81-40-21) dated February 12, 1981.

b. II.K.1 (10) & (23) - IE Bulletin 79-08

Commonwealth Edison has performed a comprehensive review of the procedures associated with operability status (II.K.1(10) and reactor vessel water level (II.K.1. (23)). This review was documented and provided to the Regional Office of Inspection and Enforcement (Resident Inspector). It is judged, therefore, that LaSalle County Station is in conformance with this task action item.

c. II.K.3-B & O Task Force Recommendations

Commonwealth Edison has committed to provide additional information in support of the following responses now documented in the FSAR:

(i) L.34-13 "HPCS Auto Restart" (II.K.3 (21))
(ii) L.34-16 "Emergency Power on Pump Seals" (II.K.3 (25))
(iii) L.34.24 "ACRS Consultant Questions" (II.K.3.45)

With respect to item (i), the Staff expressed concern that the HPCS system, though reviewed as a part of the BWR Owners analysis of this task, was not addressed explicitly in the conclusions stated for LaSalle County in Section L.34-13 of the FSAR. The FSAR has been reviewed and appropriate changes made to resolve this concern. The proposed change to L.34-13 is attached and will be submitted formally in a future amendment to the FSAR. Also worthy of note is the fact that the LaSalle HPCS will auto-restart after a manual termination.

With respect to item (ii), the Commonwealth Edison design does not now provide emergency power to the recirculation pump seal cooling water. Therefore, the current position stated in L.34-16 of the FSAR can not yet be changed. However, it is worth noting that these cooling water systems can be manually transferred to an emergency bus if required.

With respect to item (iii), the Commonwealth Edison review is expected to be completed before February 18, 1981. The results of that review will be transmitted immediately upon completion.

Two other items are discussed here only to re-emphasize the point that although commitments have been made to modify the plant as recommended in NUREG-0737, equipment delivery schedules are expected to preclude satisfying the required implementation dates. The two items in question are:

(i) RCIC Auto Restart (L.34-6, II.K.3(13))

Availability of qualified components will prohibit final implementation by 1/1/81. System design for this modification was initiated in the Fall, 1980 prior to imposition of the requirement in November, 1980. Final design was completed in December, 1980 and equipment purchases will be expedited to assure installation at the first outage of sufficient duration to complete the work.

(ii) RCIC Suction Transfer (L.34-14, II.K.3(22))

The situation for this modification is the same as described for the RCIC auto-restart. For that reason, it is judged that the implementation dated 1/1/81 will not be met.

For both items II.K.3.13 and II.K.3.22, equipment delivery schedules can be provided to the Staff upon accentance of firm proposals from equipment suppliers. These schedules would reflect the impact of equipment qualification testing as well.

Item II.K.3.21 have been reviewed. This review has included a consideration of all aspects of HPCS, LPCS, and LPCI system operations which would be influenced by any expanded automatic restart capability. It is concluded that the current system design is adequate and no design changes are required. This conclusion is based on a combination of factors that include: the comprehensive nature of BWR operator training, the emphasis placed in this training on reactor water level control, the Emergency Procedure Guidelines, the relatively long time the operator has to correct errors and the extent to which low reactor water level conditions are displayed and alarmed in the control room. The most important consideration is that the benefits of providing enhanced automatic ECCS reinitiation do not justify the associated penalties of increased system complexity, reduced system reliability, restricted operator flexibility and the other undesirable effects discussed in this memorandum.

The NSSS Vendor and the BWR Owners' Group believe the current BWR ECCS design, when coupled with rigorous and continuous operating staff training programs, represents the optimum approach to BWR safety. No modifications of existing LPCI ~~and~~ LPCS AND HPCS need to be undertaken.

2. One of the factors which supports the current adequacy conclusion is the period of grace-time available to the operator between the instant when the operator should have started an idle ECC system (but does not) and the instant when predefined core cooling difficulties would occur (fuel clad attains 2200° F).

Given a BWR core initially at saturation temperature conditions without any source of makeup reactor water (because the operator has erroneously terminated ECCS pump flow), the following tabulation summarizes the time interval between pump flow termination and occurrence of 2200° F fuel clad temperature:

Postulated Cases	Minimum Time Interval To Reach 2200° F
Case 1. Boil off from Level 1 31.5 inches above top of active fuel	30 minutes
Case 2. Boil off from top of jet pump	15 to 20 minutes

In Case 1, the reactor water level is initially at the ECCS initiation value (Level 1). It is assumed that there is no ECCS flow and the reactor boil-off process results in decreasing reactor water level leading eventually to core