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NUCLEAR REGULATORY COMMISSION
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February 18, 1981

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Docket Nos. 50-373
and 50-374

MEMORANDUM FOR: B. J. Youngblood, Chief
Licensing Branch #1, DL

FROM: Karl Kniel, Chief
Generic Issues Branch, DST

SUBJECT: SER INPUT - LASALLE UNIT NOS. 1 AND 2

Plant Name: LaSalle Unit Nos. 1 and 2
Docket Numbers: 50-373 and 50-374
Licensing Stage: OL
Responsible Branch and Project Manager: LB#1, A. Bournia
DST Branch Involved: Generic Issues Branch
Description of Review: Unresolved Safety Issues
Requested Completion Date: February 18, 1981
Review Status: Complete

The Generic Issues Branch, DST, input to the LaSalle Unit Nos. 1 and 2 Safety Evaluation Report is enclosed. This appendix to the SER addresses the status of Unresolved Safety Issues pertaining to these facilities, and is in response to the ALAB-444 decision on this subject. That decision specified that "...each SER should contain a summary description of those generic problems under continuing study which have both relevance to facilities of the type under review and potentially significant public safety implications."

Karl Kniel

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Input to SER

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APPENDIX A

NUCLEAR REGULATORY COMMISSION UNRESOLVED SAFETY ISSUES

A.1 Unresolved Safety Issues

The NRC staff continuously evaluates the safety requirements used in its reviews against new information as it becomes available. Information related to the safety of nuclear power plants comes from a variety of sources including experience from operating reactors; research results, NRC staff and Advisory Committee on Reactor Safeguards safety reviews; and vendor, architect/engineer and utility design reviews. Each time a new concern or safety issue is identified from one or more of these sources, the need for immediate action to assure safe operation is assessed. This assessment includes consideration of the generic implications of the issue.

In some cases, immediate action is taken to assure safety, e.g., the derating of boiling water reactors as a result of the channel box wear problems in 1975. In other cases, interim measures, such as modifications to operating procedures, may be sufficient to allow further study of the issue prior to making licensing decisions.

In most cases, however, the initial assessment indicates that immediate licensing actions or changes in licensing criteria are not necessary. In any event, further study may be deemed appropriate to make judgments as to whether existing NRC staff requirements should be modified to address the issue for new plants or if backfitting is appropriate for the long-term operation of plants already under construction or in operation.

These issues are sometimes called "generic safety issues" because they are related to a particular class or type of nuclear facility rather than a specific plant. Certain of these issues have been designated as "unresolved safety issues."(1) However, as discussed above, such issues are considered on a generic basis only after the staff has made an initial determination that the safety significance of the issue does not prohibit continued operation or require licensing actions while the longer-term generic review is underway.

A.2 ALAB-444 Requirements

These longer-term generic studies were the subject of a Decision by the Atomic Safety and Licensing Appeal Board of the Nuclear Regulatory Commission. The Decision was issued on November 23, 1977 (ALAB-444) in connection with the Appeal Board's consideration of the Gulf States Utility Company application for the River Bend Station, Unit Nos. 1 and 2.

In the view of the Appeal Board (pp. 25-29):

"The responsibilities of a licensing board in the radiological health and safety sphere are not confined to the consideration and disposition of those issues which may have been presented to it by a party or an "Interested State" with the required degree of specificity. To the contrary, irrespective of what matters may or may not have been properly placed in controversy, prior to authorizing the issuance of a construction permit the board must make the finding, interalia, that there is "reasonable assurance" that "the proposed facility can be constructed and operated at the proposed location without undue risk to the health and safety of the public." Of necessity, this 10 CFR 50.35(a) determination will entail an inquiry into whether the staff review satisfactorily has come to the grips with any unresolved generic safety problems which might have an impact upon operation of the nuclear facility under consideration."

"The SER is, of course, the principal document before the licensing board which reflects the content and outcome of the staff's safety review. The board should therefore be able to look to that document to ascertain the extent to which generic unresolved safety problems which have been previously identified in a FSAR item, a Task Action Plan, an ACRS report or elsewhere have been factored into the staff's analysis for the particular reactor -- and with what result. To this end, in our view, each SER should contain a summary description of generic problems under continuing study which have both rele to facilities of the type under review and potentially signi public safety implications."

"This summary description should include information of the kind now contained in most Task Action Plans. More specifically, there should be an indication of the investigative program which has been or will be undertaken with regard to the problem, the program's anticipated time span, whether (and if so, what) interim measures have been devised for dealing with the problem pending the completion of the investigation, and what alternative courses of action might be available should the program not produce the envisaged result."

"In short, the board (and the public as well) should be in a position to ascertain from the SER itself -- without the need to resort to extrinsic documents -- the staff's perception of the nature and extent of the relationship between each significant unresolved generic safety question and the eventual operation of the reactor under scrutiny. Once again, this assessment might well have a direct bearing upon the ability of the licensing board to make the safety findings required of it on the construction permit level even though the generic answer to the question remains in the offing. Among other things, the furnished information likely shed light on such alternatively important considerations as whether: (1) the problem has already been resolved for the reactor under study; (2) there is a reasonable basis for concluding that a satisfactory solution will be obtained before the reactor is put in operation; or (3) the problem would have no safety implications until after several years of reactor operation and, should it not be resolved by then, alternative means will be available to insure that continued operation (if permitted at all) would not pose an undue risk to the public."

This appendix is specifically included to respond to the decision of the Atomic Safety and Licensing Appeal Board as enunciated in ALAB-444 and as applied to an operating license proceeding involving Virginia Electric and Power Company (North Anna Nuclear Power Station, Units 1 and 2), ALAB-491, NRC 245 (1978).

A.3 "Unresolved Safety Issues"

In a related matter, as a result of Congressional action on the Nuclear Regulatory Commission budget for Fiscal Year 1978, the Energy Reorganization Act of 1974 was amended (PL 95-209) on December 13, 1977 to include, among other things, a new Section 210 as follows:

"UNRESOLVED SAFETY ISSUES PLAN"

"SEC. 210. The Commission shall develop a plan providing for specification and analysis of unresolved safety issues relating to nuclear reactors and shall take such action as may be necessary to implement corrective measures with respect to such issues. Such plan shall be submitted to the Congress on or before January 1, 1978 and progress reports shall be included in the annual report of the Commission thereafter."

The Joint Explanatory Statement of the House-Senate Conference Committee for the FY 1978 Appropriations Bill (Bill S.1131) provided the following additional information regarding the Committee's deliberations on this portion of the bill:

"SECTION 3 - UNRESOLVED SAFETY ISSUES"

"The House amendment required development of a plan to resolve generic safety issues. The conferees agreed to a requirement that the plan be submitted to the Congress on or before January 1, 1978. The conferees also expressed the intent that this plan should identify and describe those safety issues, relating to nuclear power reactors, which are unresolved on the date of enactment. It should set forth: (1) Commission actions taken directly or indirectly to develop and implement corrective measures; (2) further actions planned concerning such measures; and (3) timetables and cost estimates of such actions. The Commission should indicate the priority it has assigned to each issue, and the basis on which priorities have been assigned."

In response to the reporting requirements of the new Section 210, the NRC staff submitted to Congress on January 1, 1978, a report describing the NRC generic issues program (NUREG-0410)[1/]. The NRC program was already in place when PL 95-209 was enacted and is of considerably broader scope than the "Unresolved Safety Issues Plan" required by Section 210. In the letter transmitting NUREG-0410 to the Congress on December 30, 1977, the Commission indicated that "the progress reports, which are required by Section 210 to be included in future NRC annual reports, may be more useful to Congress if they focus on the specific Section 210 safety items.

It is the NRC's view that the intent of Section 210 was to assure that plans were developed and implemented on issues with potentially significant public safety implications. In 1978, the NRC undertook a review of over 130 generic issues addressed in the NRC program to determine which issues fit this description and qualify as "Unresolved Safety Issues" for reporting to the Congress. The NRC review included the development of proposals by the NRC Staff and review and final approval by the NRC Commissioners.

This review is described in a report, NUREG-0510, entitled "Identification of Unresolved Safety Issues Relating to Nuclear Power Plants - A Report to Congress" dated January 1979. The report provides the following definition of an "Unresolved Safety Issue."

"An Unresolved Safety Issue is a matter affecting a number of nuclear power plants that poses important questions concerning the adequacy of existing safety requirements for which the final resolution has not yet been developed and that involves conditions not likely to be acceptable over the lifetime of the plants it affects."

Further, the report indicates that in applying this definition, matters that pose "important questions concerning the adequacy of existing safety requirements" were judged to be those for which resolution is necessary to (1) compensate for a possible major reduction in the degree of protection of the public health and safety, or (2) provide a potentially significant decrease in the risk to the public health and safety. Quite simply, an "Unresolved Safety Issue" is potentially significant from a public safety standpoint and its resolution is likely to result in NRC action on the affected plants.

All of the issues addressed in the NRC program were systematically evaluated against this definition as described in NUREG-0510. As a result, 17 "Unresolved Safety Issues" addressed by 22 tasks in the NRC program were identified. The issues are listed below. Progress on these issues was first discussed in the 1978 NRC Annual Report. The number(s) of the generic task(s) (e.g., A-1) in the NRC program addressing each issue is indicated in parentheses following the title.

<u>"UNRESOLVED SAFETY ISSUES" (APPLICABLE TASK NOS.)</u>

1. Water Hammer - (A-1)
2. Asymmetric Blowdown Loads on the Reactor Coolant System - (A-2)
3. Pressurized Water Reactor Steam Generator Tube Integrity - (A-3, A-4, A-5)

4. BWR Mark I and Mark II Pressure Suppression Containments - (A-6, A-7, A-8, A-39)

5. Anticipated Transients Without Scram - (A-9)

6. BWR Nozzle Cracking - (A-10)

7. Reactor Vessel Materials Toughness - (A-11)

8. Fracture Toughness of Steam Generator and Reactor Coolant Pump Supports - (A-12)

9. Systems Interaction in Nuclear Power Plants - (A-17)

10. Environmental Qualification of Safety-Related Electrical Equipment - (A-24)

11. Reactor Vessel Pressure Transient Protection - (A-26)

12. Residual Heat Removal Requirements - (A-31)

13. Control of Heavy Loads Near Spent Fuel - (A-36)

14. Seismic Design Criteria - (A-40)

15. Pipe Cracks at Boiling Water Reactors - (A-42)

16. Containment Emergency Sump Reliability - (A-43)

17. Station Blackout - (A-44)

In the view of the staff, the "Unresolved Safety Issues" listed above are the substantive safety issues referred to by the Appeal Board in ALAB-444 when it spoke of "...those generic problems continuing study which have ... potentially significant public safety implications" (page 27). Six of the 22 tasks identified with the "Unresolved Safety Issues" are not applicable to LaSalle Units 1 and 2 because they apply to pressurized water reactors only. These tasks are A-2, A-3, A-4, A-5, A-12, and A-26.

With regard to the 16 remaining tasks that are applicable to LaSalle Units 1 and 2, the NRC staff has issued NUREG reports providing its proposed resolution of seven of the issues.

The table below lists those issues.

Task No.	NUREG Report and Title	SER/SER Suppl. Section
A-8	NUREG-0487, "Mark II Containment Lead Plant Program Load Evaluation and Acceptance Criteria." October 1978. Supplement 1 to NUREG-0487, October 1980. Supplement 2 to NUREG-0487, February 1981.	6.2.1.1
A-24	NUREG-0588, "Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment."	3.10
A-26	NUREG-0224, "Reactor Vessel Pressure Transient Protection for Pressurized Water Reactors" and RSB BTP 5-2.	5.2.2
A-31	SRP 5.4.7 and BTP 5-1 "Residual Heat Removal Systems" incorporate requirements of USI A-31.	5.4.7
A-36	NUREG-0612, "Control of Heavy Loads at Nuclear Powers Plants"	9.1.2, 9.1.4
A-39	NUREG-0487 and Supplement 1 to NUREG-0487 (See above).	6.2.1.1
A-42	NUREG-0313, Rev. 1	5.2.3

The remaining issues applicable to LaSalle Units 1 and 2 are listed in the following table.

GENERIC TASKS ADDRESSING UNRESOLVED
SAFETY ISSUES THAT ARE APPLICABLE TO THE
LASALLE NUCLEAR STATION, UNITS 1 AND 2

1. A-1 Water Hammer
2. A-9 ATWS
3. A-10 BWR Nozzle Cracking
4. A-11 Reactor Vessel Material Toughness
5. A-17 Systems Interactions in Nuclear Power Plants
6. A-40 Seismic Design Criteria
7. A-43 Containment Emergency Sump Reliability
8. A-44 Station Blackout

With the exception of Tasks A-9, A-43 and A-44, Task Action Plans for the generic tasks above are included in NUREG-0649, "Task Action Plans for Unresolved Safety Issues Related to Nuclear Power Plants." A technical resolution for Task A-9 has been proposed by the NRC staff in Volume 4 of NUREG-0460, issued for comment. This served as a basis for the staff's proposal for rulemaking on this issue. The Task Action Plan for Task A-43 was issued in January 1981, and the Task Action Plan for A-44 was issued in July 1980. The information provided in NUREG-0649 meets most of the informational requirements of ALAB-444. Each Task Action Plan provides a description of the problem; the staff's approaches to its resolution; a general discussion of the bases upon which continued plant licensing or operation can proceed pending completion of the task; the technical organizations involved in the task and estimates of the manpower required; a description of the interactions with other NRC offices, the Advisory Committee on Reactor Safeguards and outside organizations;

estimates of funding required for contractor supplied technical assistance; prospective dates for completing the task; and a description of potential problems that could alter the planned approach or schedule.

We have reviewed the 8 "Unresolved Safety Issues" listed above and the four new USIs discussed in Section A.4 as they relate to LaSalle Unit 1. Discussion of each of thse issues including references to related discussions in the Safety Evaluation Report is provided below in Section A.5. Based on our review of these items, we have concluded, for the reasons set forth in Section A-5, that there is reasonable assurance that the LaSalle Nuclear Station Unit 1 can be operated prior to the ultimate resolution of these generic issues without endangering the health and safety of the public.

A.4 New "Unresolved Safety Issues"

An in-depth and systematic review of generic safety concerns identified since 1979 has been performed by the staff, and resulted in a proposed list of several new "Unresolved Safety Issues." This proposed list was contained in a staff paper to the Commission, SECY 80-325 and supplemented by a memo of September 10, 1980 and SECY 80-325A. The candidate issues originated fron concerns identified in the TMI Action Plan (NUREG-0660), ACRS recommendations, abnormal occurrence reports and other operating experience. The staff's proposed list has been reviewed and commented on by the ACRS, the Office of Analysis and Evaluation of Operational Data

(AEOD), and the Office of Policy Evaluation (OPE). The ACRS and AEOD also proposed that several additional USIs be considered by the Commission. The Commission considered the above information and approved* only the following four new USIs.

A-45 Shutdown Decay Heat Removal Requirements

A-46 Seismic Qualification of Equipment in Operating Plants

A-47 Safety Implication of Control Systems (including steam generator and reactor overfill transients)

A-48 Hydrogen Control Measures and Effects of Hydrogen Burns on Safety Equipment

The applicability and bases for licensing prior to ultimate resolution of the four new USIs for LaSalle Units 1 and 2 are also discussed in Section A.5.

A.5 Discussion of Tasks as they Relate to LaSalle Units 1 and 2

A-1 Waterhammer

Waterhammer events are intense pressure pulses in fluid systems caused by any one of a number of mechanisms and system conditions such as rapid condensation of steam pockets, steam-driven slugs of water, pump startup with partially empty lines, and rapid valve motion. Since 1971, over 200 incidents involving waterhammer in pressurized boiling water reactors have been reported.

*Letter, S. J. Chilk, to W. J. Dircks, Subject: SECY-80-325 - Special Report to Congress Identifying Unresolved Safety Issues (Commission Action Item), dated December 24, 1980.

The waterhammers (or steam hammers) have involved steam generator feedrings and piping, the residual heat removal system, emergency core cooling systems, and containment spray, service water, feedwater and steam lines.

Most of the damage reported has been relatively minor, involving pipe hangers and restraints; however, several waterhammer incidents have resulted in piping and valve damage.

The most serious waterhammer events have occurred in the steam generator feedrings of pressurized water reactors. In no case has any waterhammer incident resulted in the release of radioactive material.

Under Generic Task A-1, the potential for waterhammer in various systems is being evaluated and appropriate requirements and systematic review procedures are being developed to ensure that waterhammer is given appropriate consideration in all areas of licensing review. A technical report, NUREG-0582, "Water Hammer in Nuclear Power Plants," (July 1979) providing the results of an NRC staff review of waterhammer events in nuclear power plants and stating current staff licensing positions, completes a major subtask of Generic Task A-1.

Although waterhammer can occur in any LWR and approxiamtely 118 actual and probable events have been reported in BWRs as of September 1979, none have caused major pipe failures in a BWR such as LaSalle and none have resulted in the offsite release of radioactivity.

LaSalle has installed a system to preclude waterhammer from occurring in ECCS lines. This system consists of jockey pumps to keep ECCS lines water-filled so that ECCS pumps will not start pumping into voided lines and steam will not collect in the ECCS piping. To ensure that the ECCS lines remain water-filled, vents have been installed and a Tech Spec requirement to periodically vent air from the lines has been imposed.

With regard to additional protection against potential waterhammer events currently provided in plants, piping design codes require consideration of impact loads. Approaches used at the design stage include: (1) increasing valve closure times, (2) piping layout to preclude water slugs in steam lines and vapor formation in water lines, (3) use of snubbers and pipe hangers, and (4) use of vents and drains. In addition, we require that applicants conduct a preoperational vibration dynamic effects test program in accordance with Section III of the ASME Code for all ASME Class 1 and Class 2 piping systems and piping restraints during startup and initial operation. These tests will provide adequate assurance that the piping and piping restraints have been designed to withstand dynamic effects due to valve closures, pump trips and other operating modes associated with the design operational transients.

Nonetheless, in the unlikely event that a large pipe break did result from a severe waterhammer event, core cooling is assured by the emergency core cooling systems and protection against the dynamic effects of such pipe breaks inside and outside of containment is provided.

In the event that Task A-1 identifies potentially significant waterhammer scenarios that have not explicitly been accounted for in the design and operation of the LaSalle Unit, corrective measures will be required at that time. The task has not identified the need for measures beyond those already implemented.

Based on the foregoing, we have concluded that LaSalle Unit 1 can be operated prior to resolution of this generic issue without undue risk to the health and safety of the public.

A-9 Anticipated Transients Without Scram (ATWS)

Nuclear plants have safety and control systems to limit the consequences of temporary abnormal operating conditions or "anticipated transients." Some deviations from normal operating conditions may be minor; others, occurring less frequently, may impose significant demands on plant equipment. In some anticipated transients, rapidly shutting down the nuclear reaction (initiating a "scram"), and thus rapidly reducing the generation of heat in the reactor core, is an important safety measure. If there were a potentially severe "anticipated transient" and the reactor shutdown system did not "scram" as desired, then an "anticipated transient without scram," or ATWS, would have occurred.

All BWRs including LaSalle have been required to provide recirculation pump trip in the event of a reactor trip and to provide additional operator training for recovery from ATWS events.

A recirculation pump trip provision has been incorporated in the LaSalle design. In addition, emergency procedures and operator training have been implemented to cope with potential ATWS events.

Operator training and action as described, in conjunction with the automatic recirculation pump trip, significantly improves the capability of the facility to withstand a range of ATWS events, such that operation of this facility presents no undue risk to the health and safety of the public while this matter is under review.

The ATWS issue is currently scheduled for rulemaking in mid-summer 1981. The applicant will be required to comply with any further requirements on ATWS which may be imposed as a result of the rulemaking.

Based on our review, we have concluded that there is reasonable assurance that LaSalle can be operated prior to ultimate resolution of this generic issue without endangering the health and safety of the public.

A-10 BWR Nozzle Cracking

Cracks have been found in the feedwater nozzles of essentially all operating BWRs. These cracks have been found in both the blend radius and bore regions and in many cases have penetrated the stainless steel cladding into the carbon steel base metal.

Cracking problems have been discovered on BWR control rod drive return line nozzles, which are also stainless steel clad and through which cold water flows continuously.

This generic issue has been under review by the NRC staff since 1977 and has been technically resolved. Required preventive measures have been outlined in NUREG-0619 which was issued for comment in April 1980. The final version of NUREG-0619 will be issued in early 1981. All applicants including LaSalle are required to comply with the implementative measures specified in the NUREG document.

LaSalle Unit 1 feedwater spargers are stainless steel headers with six headers served through six feedwater nozzles, each fitted with triple thermal sleeves. This design is in conformance with requirements of NUREG-0619.

The LaSalle Unit 1 Control Rod Drive Return Nozzle has been capped and the line eliminated. This modification satisfies the requirements of NUREG-0619 in that it eliminates the thermal cycling problem identified for these nozzles.

Based on the foregoing, we conclude that LaSalle has complied with the requirements of NUREG-0619 and therefore can be operated without undue risk to the health and safety of the public.

A-11 Reactor Vessel Materials Toughness

Resistance to brittle fracture is described quantitatively by a material property generally denoted as "fracture toughness."

Fracture toughness has different values and characteristics depending upon the material being considered. For steels used in nuclear reactor pressure vessels, three considerations are important. First, fracture toughness increases with increasing temperature. Second, fracture toughness decreases with increasing load rates. Third, fracture toughness decreases with neutron irradiation.

In recognition of these conditions, power reactors are operated within restrictions imposed by Technical Specifications on pressure during heatup and cooldown operations. These restrictions assure that the reactor vessel will not be subjected to a combination of pressure and temperature that could cause brittle fracture of the vessel if there were significant flaws in the vessel material. The effect of neutron radiation on fracture toughness of the vessel material over the life of the plant is accounted for in Technical Specification limitations.

The principal objective of Task A-11 is to develop safety criteria to allow a more precise assessment of safety margins during normal operation, transients and accident conditions in older reactor vessels with marginal fracture toughness.

Based on our evaluation of the LaSalle reactor vessel materials toughness, we have concluded that adequate safety margins exist for brittle failure during operating, testing, maintenance and anticipated transient conditions. When Task A-11 is completed and explicit fracture evaluation criteria for accident conditions are defined, all vessels will be reevaluated for acceptability over their design lives.

Based on minimum acceptable charpy impact values of 20 ft. lbs., and fabrication techniques employed on the LaSalle vessel, we conservatively estimate that the total fluence over the design life would result in end of fracture toughness above the minimum charpy impact requirement of 50 ft. lbs. In addition, the surveillance program required by 10 CFR 50, Appendix H will afford an opportunity to reevaluate the fracture toughness periodically during the first half of design life.

Therefore, based upon the foregoing, we have concluded that LaSalle can be operated prior to resolution of this generic issue without undue risk to the health and safety of the public.

A-17 Systems Interactions in Nuclear Power Plants

The licensing requirements and procedures used in our safety review address many different types of systems interactions. Current licensing requirements are founded on the principle of defense-in-depth. Adherence to this principle results in requirements such as physical separation and independence of redundant safety systems, and protection against events such as high energy line ruptures, missiles, high winds, flooding, seismic events, fires, operator errors, and sabotage. These design provisions supplemented by the current review procedures of the Standard Review Plan (NUREG-75/087) which require interdisciplinary reviews and which account, to a large extent, for review of potential systems interactions, provide for an adequately safe situation with respect to such interactions. The quality assurance program which is followed during the design, construction, and operational phases for each plant is expected to provide added assurance against the potential for adverse systems interactions.

In November 1974, the Advisory Committee on Reactor Safeguards requested that the NRC staff give attention to the evaluation of safety systems from a multi-disciplinary point of view, in order to identify potentially undesirable interactions between plant systems. The concern arises because the design and analysis of systems is frequently assigned to teams with functional engineering specialities--such as civil, electrical, mechanical, or nuclear. The question is whether the work of these function specialists is sufficiently integrated in their design and analysis activities to enable them to identify adverse interactions between and among systems. Such adverse events might occur, for example, because designers did not assure that redundancy and independence of safety systems were provided under all conditions of operation required, which might happen if the function teams were not adequately coordinated.

In mid-1977, Task A-17 was initiated to confirm that present review procedures and safety criteria provide an acceptable level of redundancy and independence for systems required for safety by evaluating the potential for undesirable interactions between and among systems.

The NRC staff's current review procedures assign primary responsibility for review of various technical areas and safety systems to specific organization units and assign secondary responsibility to other units where there is a function or interdisciplinary relationship. Designers follow somewhat similar procedures and provide for interdisciplinary reviews and analyses of systems. Task A-17 provided an independent study of m[illegible]hat could identify important systems interactions adversely impacting safety; and which are not considered by current review procedures. The first phase of this study began in May 1978, and was completed in February 1980 by Sandia Laboratories under contract to the NRC staff.

The Phase 1 investigation was structured to identify areas where interactions are possible between and among systems and have the potential of negating or seriously degrading the performance of safety functions. The study concentrated on common cause or linking failures among systems that could violate a safety function. The investigation then identified where NRC review procedures may not have properly accounted for these interactions.

The Sandia Study used fault-tree methods to identify component failure combinations (cut-sets) that could result in loss of a safety function. The cut-sets were reduced to minimal combinations by incorporating six common or linking systems failures into the analysis. The results of the Phase 1 effort indicate that, within the scope of the study, only a few areas of review procedures need improvement regarding systems interaction. However, the level of detail needed to identify all examples of potential system interaction candidates observed in some operating plants was not within the Phase 1 scope of the Sandia Study.

The Systems Interaction Branch formed in NRR in April 1980, has been studying state-of-the-art methods that can be used to predict systems interactions. The initial effort, supported by three laboratory contractors, is underway; a range of methods is being considered and tested for feasibility against a sample of some systems interaction candidates derived from Licensee Event Report evaluations.

It is expected that the development of systematic ways to identify and evaluate systems interactions will reduce the likelihood of common cause failures resulting in the loss of plant safety functions. However, the studies to date indicate that current review procedures and criteria supplemented by the application of post-TMI findings and risk studies provide reasonable assurance that the effects of potential systems interaction on plant safety will be within the effects on plant safety previously evaluated.

Therefore, we concluded that there is reasonable assurance that LaSalle Unit 1 can be operated prior to the final resolution of this generic issue without endangering the health and safety of the public.

A-40 Seismic Design Criteria - Short-Term Program

NRC regulations require that nuclear power structures, systems and components important to safety be designed to withstand the effects of natural phenomena such as earthquakes. Detailed requirements and guidance regarding the seismic design of nuclear plants are provided in the NRC regulations and in Regulatory Guides issued by the Commission. However, there are a number of plants with construction permits and operating licenses issued before the NRC's current regulations and regulatory guidance were in place. For this reason, rereviews of the seismic design of various plants are being undertaken to assure that these plants do not present an undue risk to the public. Task A-40 is, in effect, a compendium of short-term efforts to support such reevaluation efforts of the NRC staff, especially those related to older operating plants. In addition, some revisions to SRP sections and Regulatory Guides to bring them more in line with the state-of-the-art will result.

The seismic design basis and seismic design of LaSalle Unit 1 has been evaluated at the operating license stage and have been found acceptable. Seismic design review of LaSalle was conducted using current licensing criteria and requirements. Should the resolution of Task A-40 indicate a change is needed in these licensing requirements, all operating reactors, including LaSalle will be re-evaluated on a case-by-case basis. Accordingly, we have concluded that LaSalle Unit 1 can be operated prior to ultimate resolution of this generic issue without endangering the health and safety of the public.

A-43 Containment Emergency Sump Reliability

Following a postulated loss of coolant accident, i.e., a break in the reactor coolant system piping, the water flowing from the break would be collected in the suppression pool. This water would be recirculated through the reactor system by the emergency core cooling pumps to maintain core cooling. This water may also be circulated through the containment spray system to remove heat and fission products from the drywell and wetwell atmosphere. Loss of the ability to draw water from the suppression pool could disable the emergency core cooling and containment spray systems.

The concern addressed by this Task Action Plan for Boiling Water Reactors (BWR) is limited to the potential for degraded ECCS performance as a result of thermal insulation debris that may be blown into the suppression pool during a loss-of-coolant accident and cause blockage of the pump suction lines. A second concern, potential vortex formation, is not considered a serious concern for Mark II containment due to the large depth of the pool (approximately 25 feet) and the low approach velocities (see Section 6.3.2).

With regard to potential blockage of the intake lines, the likelihood of any insulation being drawn into an ECCS pump suction line is very small. The potential debris in the drywell could only be swept into the suppression pool via the downcomer piping. However, the downcomer pipes (2 foot diamter) are capped with jet deflectors and would prevent any large pieces from reaching the suppression pool. Any smaller pieces reaching the pool would tend to settle on the bottom and would not be drawn into the pump suction since it is located several feet above the pool bottom. In addition, BWR designs employ strainers within the suction piping and NPSH calculations for the pumps are based on an assumed blockage of 50%. Accordingly, we have concluded that LaSalle can be operated prior to ultimate resolution of this generic issue without endangering the health and safety of the public.

A-44 Station Blackout

Electrical power for safety systems at nuclear power plants must be supplied by at least two redundant and independent divisions. The systems used to remove decay heat to cool the reactor core following a reactor shutdown are included among the safety systems that must meet these requirements. Each electrical division for safety systems includes an offsite alternating current (ac) power connection, a standby emergency diesel generator ac power supply, and direct current (dc) sources.

Task A-44 involves a study of whether or not nuclear power plants should be designed to accommodate a complete loss of all ac power, i.e., a loss of both the offsite and the emergency diesel generator ac power supplies. This issue arose because of operating experience regarding the reliability of ac power supplies. A number of operating plants have experienced a total loss of offsite electrical power, and more occurrences are expected in the future. During each of these loss of offsite power events, the onsite emergency ac power supplies were available to supply the power needed by vital safety equipment. However, in some instances, one of the redundant emergency power supplies has been unavailable. In addition, there have been numerous reports of emergency diesel generators failing to start and run in operating plants during periodic surveillance tests.

A loss of all ac power was not a design basis event for the LaSalle facility. Nonetheless, a combination of design, operation and testing requirements that have been imposed on the applicant will assure that these units will have substantial resistance to a loss of all ac and that, even if a loss of all ac should occur, there is reasonable assurance that the core will be cooled. These are discussed below.

A loss of offsite ac power involves a loss of both the preferred and backup sources of offsite power. Our review and basis for acceptance of the design, inspection, and testing provisions for the offsite power system are described in Section 8.2 of the LaSalle SER.

If offsite ac power is lost, three diesel generators and their associated distribution systems will deliver emergency power to safety-related equipment. Our review of the design, testing, surveillance, and maintenance provisions for the onsite emergency diesels is described in Section 8.3 of the SER. Our requirements include preoperational testing to assure the reliability of the installed diesel generators in accordance with our requirements discussed in the SER. In addition, the applicant has been requested to implement a program for enhancement of diesel generator reliability to better assure the long-term reliability of the diesel generators.

If both offsite and onsite ac power are lost, boiling water reactors may use a combination of safety/relief valves and the reactor core isolation cooling system (RCIC) to remove decay heat without reliance on ac power. These systems assure that adequate cooling can be maintained for at least two hours, which allows time for restoration of ac power from either offsite or onsite sources.

The issue of station blackout was also considered by the Atomic Safety and Licensing Appeal Board (ALAB-603) for the St. Lucie Unit No. 2 facility. In addition, in view of the completion schedule for Task A-44 (October 1982), the Appeal Board recommended that the Commission take expeditious action to ensure that other plants and their operators are equipped to accommodate a station blackout event. The Commission has reviewed this recommendation and determined that some interim measures should be taken at all facilities including LaSalle while Task A-44 is being conducted. Consequently interim emergency procedures and operator training for safe operation of the facility and restoration of AC power will be required. This action is required to be completed by fuel load date.

Based on the above, we have concluded that there is reasonable assurance that LaSalle Unit No. 1 can be operated prior to the ultimate resolution of this generic issue without endangering the health and safety of the public.

A-45 Shutdown Decay Heat Removal Requirements

Following a reactor shutdown, the radioactive decay of fission products continues to produce heat (decay heat) which must be removed from the primary system. The principal means for removing this heat in a boiling water reactor while at high pressure is via the steam lines to the turbine condenser. The condensate is normally returned to the reactor vessel by the feedwater system, however, the steam turbine driven reactor core isolation cooling system (RCIC) is provided to maintain primary system inventory, if AC power is not available. When the system is at low pressure, the decay heat is removed by the residual heat removal systems (RHR). This USI will evaluate the benefit of providing alternate means of decay heat removal which could substantially increase the plants' capability to handle a broader spectrum of transients and accidents. The study will consist of a generic system evaluation and will result in recommendations regarding the desirability of and possible design requirements for improvements in existing systems or an alternative decay heat removal method if the improvements or alternative can significantly reduce the overall risk to the public.

The LaSalle reactors have various methods for the removal of decay heat. As discussed above, the decay heat is normally rejected to the turbine condenser and returned to the vessel by either the feedwater system or the RCIC (from the condensate storage tank). If the condenser is not available (e.g., loss of offsite power), heat can be removed via the safety/relief valves to the suppression pool. Also, the high pressure core spray (HPCS) system is provided if the RCIC is not available. Both of these systems can recirculate fluid to the vessel from either the condensate storage tank or the suppression pool. If the RCIC and HPCS are unavailable, the reactor system pressure can be reduced by the automatic depressurization system (ADS) so that cooling by the RHR can be initiated. When the condenser is not used, the heat rejected to the suppression pool is subsequently removed by the RHR.

The RCIC and HPCS systems at LaSalle have improvements over comparable systems at older BWRs. The RCIC has been upgraded to safety grade quality (now required for all BWRs), and the HPCS is powered by its own dedicated diesel so it can operate with an assumed loss of all other sources of AC power. Also, the RHR contains three pumps; the flow capacity of any single pump is sufficient to easily remove the decay heat. Accordingly, we have concluded that LaSalle can be operated prior to ultimate resolution of this generic issue without endangering the health and safety of the public.

A-46 Seismic Qualification of Equipment in Operating Plants

The design criteria and methods for the seismic qualification of mechanical and electrical equipment in nuclear power plants have undergone significant change during the course of the commercial nuclear power program. Consequently, the margins of safety provided in existing equipment to resist seismically induced loads and perform the intended safety functions may vary considerably. The seismic qualification of the equipment in operating plants must, therefore, be reassessed to ensure the ability to bring the plant to a safe shutdown condition when subject to a seismic event. The objective of this Unresolved Safety Issue is to establish an explicit set of guidelines that could be used to judge the adequacy of the seismic qualification of mechanical and electrical equipment at all operating plants in lieu of attempting to backfit current design criteria for new plants. This guidance will concern equipment required to safely shut down the plant, as well as equipment whose function is not required for safe shutdown, but whose failure could result in adverse conditions which might impair shutdown functions.

LaSalle Unit 1 was designed using current Seismic Criteria and the design has been reviewed and approved by the Commission staff in accordance with current design criteria and methods for seismic qualification. Therefore we conclude that LaSalle Unit 1 can be operated prior to resolution of this generic issue without undue risk to the health and safety of the public.

A-47 Safety Implications of Control Systems

This issue concerns the potential for transients or accidents being made more severe as a result of control system failures or malfunctions. These failures or malfunctions may occur independently or as a result of the accident or transient under consideration. One concern is the potential for a single failure such as a loss of a power supply, short circuit, open circuit, or sensor failure to cause simultaneous malfunction of several control features. Such an occurrence would conceivably result in a transient more severe than those transients analyzed as anticipated operational occurrences. A second concern is for a postulated accident to cause control system failures which would make the accident more severe than analyzed. Accidents could conceivably cause control system failures by creating a harsh environment in the area of the control equipment or by physically damaging the control equipment. Although it is generally believed that such control system failures would not lead to serious events or result in conditions that safety systems cannot safely handle, in-depth studies have not been rigourously performed to verify this belief. The potential for an accident that would affect a particular control system, and effects of the control system failures, may differ from plant to plant. Therefore, it is not possible to develop generic answers to these concerns, but rather plant-specific reviews are required. The purpose of this USI is to define generic criteria that will be used for plant specific reviews.

The LaSalle control and safety systems have been designed with the goal of ensuring that control system failures (either single or multiple failures) will not prevent automatic or manual initiation and operation of any safety system equipment required to trip the plant or to maintain the plant in a safe shutdown condition following any "anticipated operational occurrence" or "accident." This has been accomplished by either providing independence between safety and non-safety systems or providing isolating devices between safety and non-safety systems. These devices preclude the propagation of non-safety system equipment faults such that operation of the safety system equipment is not impaired.

A systematic evaluation of the control system design, such as contemplated for this USI, has not been performed to determine whether postulated accidents could cause significant control system failures which would make the accident consequences more severe than presently analyzed. However, a wide range of bounding transients and accidents is presently analyzed to assure that the postulated events would be adequately mitigated by the safety systems. In addition, systematic reviews of safety systems have been performed with the goal of ensuring that control system failures (single or multiple) will not defeat safety system action.

A specific subtask of this USI issue will be to study the reactor overfill transient in BWRs to determine the need for preventative and/or mitigating design measures to preclude or minimize the consequences of this transient. Several early BWRs have experienced reactor vessel overfill transients with subsequent two-phase or liquid flow through the safety/relief valves. Following these early events, control grade high level trips (level 8) have been installed at most BWRs (including LaSalle) to terminate flow from the appropriate systems. These high level trips are single failure proof and periodic surveillance is required by the Technical Specifications. No overfilling events have occurred since the level 8 trips were installed.

Based on the above, we have concluded that there is reasonable assurance that LaSalle Unit No. 1 can be operated prior to the ultimate resolution of this generic issue without endangering the health and safety of the public.

A-48 Hydrogen Control Measures and Effects of Hydrogen Burns on Safety Equipment

Following a loss-of-coolant accident in a light water reactor (LWR) plant, combustible gases, principally hydrogen, may accumulate inside the primary reactor containment as a result of: (1) metal-water reaction involving the fuel element cladding; (2) the radiolytic decomposition of the water in the reactor core and the containment sump; (3) the corrosion of certain construction materials by the spray solution; and (4) any synergistic chemical, thermal and radiolytic effects of post-accident environmental conditions on containment protective coating systems and electric cable insulation.

Because of the potential for significant hydrogen generation as the result of an accident, 10 CFR Section 50.44, "Standards for combustible gas control system in light water cooled power reactors" and General Design Criteria 41, "Containment atmosphere cleanup" in Appendix A to 10 CFR Part 50 require that systems be provided to control hydrogen concentrations in the containment atmosphere following a postulated accident to ensure that containment integrity is maintained.

10 CFR Section 50.44 requires that the combustible gas control system provided be capable of handling the hydrogen generated as a result of degradation of the emergency core cooling system such that the hydrogen release is five times the amount calculated in demonstrating compliance with 10 CFR Section 50.46 or the amount

corresponding to reaction of the cladding to a depth of 0.00023 inch, whichever amount is greater.

The accident at TMI-2 on March 28, 1979 resulted in hydrogen generation well in excess of the amounts specified in 10 CFR Section 50.44. As a result of this knowledge it became apparent to NRC that specific design measures are needed for handling larger hydrogen releases, particularly for smaller, low pressure containments. As a result, the Commission determined that a rulemaking proceeding should be undertaken to define the manner and extent to which hydrogen evolution and other effects of a degraded core need to be taken into account in plant design. An advance notice of this rulemaking proceeding on degraded core issues was published in the Federal Register on October 2, 1980.

Recognizing that a number of years may be required to complete this rulemaking proceeding, a set of short-term or interim actions relative to hydrogen control requirements were developed and implemented. These interim measures were described in a second October 2, 1980 Federal Register notice. For plants with small containments (Mark I and Mark II) such as LaSalle, the interim rule specified that inerting is required to preclude hydrogen burning.

LaSalle has committed to inerting the containment building during power operation. We, therefore, conclude that LaSalle can be operated prior to resolution of this unresolved safety issue and the proposed rulemaking without undue risk to the health and safety of the public.