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Dalwyn R. Davidson
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March 25, 1982

Mr. A. Schwencer
Chief, Licensing Branch No. 2
Division of Licensing
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Perry Nuclear Power Plant
Docket Nos. 50-440; 50-441
Response to Request for
Additional Information
Regarding the Status of
Unresolved Safety Issues

Dear Mr. Schwencer:

This letter and the enclosed information is submitted in response to you letter dated November 18, 1981 concerning the status of unresolved safety issues pertaining to Perry.

Please advise us if we can be of any further assistance.

Very Truly Yours,

Dalwyn R. Davidson

Dalwyn R. Davidson
Vice President
System Engineering and Construction

DRD: mlb

cc: John Stefano
Jay Silberg, Esq.
Max Gildner

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Cleveland Electric Illuminating Company as a member of Licensing Review Group II (LRG-II) hereby endorses the attached LRG-II position paper 1-GIB, "Interim Licensing Bases Pending Resolution of Unresolved Safety Issues." In addition, the following plant specific information is provided to supplement LRG-II descriptions of the "Unresolved Safety Issues".

Issue	PNPP Supplemental Information/ FSAR Reference
A-1 Waterhammer	ECCS Discharge Line Fill System Section 6.3.2.2.5
A-9 Anticipated Transients Without Scram	No addition information/ATWS Section 15.8
A-11 Reactor Vessel Materials Toughness	No addition information/ Ref. FSAR Section 5.3
A-17 Systems Interaction	No additional information
A-39 Safety Relief Valve Hydrodynamic Loads	No additional information
A-40 Seismic Design Criteria	No additional information
A-43 Containment Emergency Sump Reliability	PNPP minimum suction submergence 10.5 feet
A-44 Station Blackout	PNPP response to generic letter 81-04 on Emergency Procedures and Training for Station Blackout Events from D. R. Davidson to D. G. Eisenhut dated May 4, 1981
A-45 Shutdown Decay Heat Removal	No additional information
A-46 Seismic Qualification of Equipment in Operating Plants	No additional information

Issue		PNPP Supplemental Information/ FSAR Reference
A-47	Safety Implications of Control Systems	No additional information Plant specific analysis will be provided in response to ICSB questions
A-48	Hydrogen Control Measures	PNPP specific information provided in response to CSB question 480.40 (copy attached)

1-GIB

INTERIM LICENSING BASES PENDING RESOLUTION OF UNRESOLVED SAFETY ISSUES

ISSUE

LRG-II plants will develop unified bases and justification for licensing and operation while the identified generic safety issues remain unresolved and provide a summary description of relevant investigative programs and interim measures pending resolution of the unresolved safety issues.

LRG-II RESPONSE

LRG-II participants have reviewed the generic issues identified in NUREG-0606, "Unresolved Safety Issues." The following information is provided for each of the applicable "Unresolved Safety Issues" as a bases for licensing prior to ultimate resolution of these issues.

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 waterhammers in pressurized and boiling water reactors have been reported. The waterhammers (or steam hammers) have involved steam generator feedrings and piping, and residual heat removal systems, 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, "Waterhammer 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 light water reactor and over 100 actual and probable events have been reported in boiling water reactors, none have caused major pipe failures in a boiling water reactor such as the LRG II plants and none have resulted in the offsite release of radioactivity. As noted above, the most severe waterhammers observed to date have been in steam generators. Since the boiling water reactor does not utilize a steam generator, these worst cases are eliminated. Furthermore, any waterhammer which may occur in feedwater or main steam piping will not impair the emergency core cooling system since all ECCS water enters the reactor vessel via five separate reactor vessel nozzles independent of the feedwater and main steam piping.

In order to protect the LRG-II plants emergency core cooling systems against the effects of waterhammer, the ECC systems are provided with jockey pumps. These jockey pumps keep the emergency core cooling system lines water-filled so that the emergency core cooling system pumps will not start pumping into voided lines and steam will not collect in the emergency core cooling system piping. To ensure that the emergency core cooling system lines remain water-filled, vents have been installed and further assurance for filled discharge piping is provided by pressure instrumentation at the piping high point. An alarm sounds in the main control room if the pressure falls below a predetermined setpoint

A-1 Waterhammer (Cont'd)

indicating difficulty maintaining a filled discharge line. Should this occur, or if an instrument becomes inoperable, the required action is identified in the Technical Specifications.

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, LRG-II participants will conduct a preoperational vibration and dynamic effects test program in accordance with Standard OM-3 of the American Society of Mechanical Engineers for all Class 1, Class 2, Class 3 and other piping systems and piping restraints during startup and initial operation. These tests will provide adequate assurance that the piping restraints have been designed to withstand dynamic effects due to valve closures, pump trips, and other operating modes.

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 system 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 which haved not explicitly been accounted for in the design and operation of LRG-II plants, corrective measures will be implemented at that time. The task has not identified the need for measures beyond those already implemented.

Based on the foregoing, we conclude that the LRG-II plants can be operated prior to ultimate resolution of this generic issue without undue risk to the health and safety of the public.

A-9 Anticipated Transients Without Scram

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.

A.-9 Anticipated Transients Without Scram (Cont'd)

All boiling water reactors, including LRG-II plants, have been required to provide recirculation pump trip in the event of a reactor trip and to provide additiona operator training for recovery from anticipated transients without scram events. In addition, LRG-II plants will implement emergency procedures and operator training to cope with potential anticipated transients without scram 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 anticipated transient without scram 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 anticipated transient without scram issue is currently under review through the rulemaking proceedings. Notice of the proposed rule for ATWS was published in the Federal Register on November 24, 1981. LRG-II plants will comply with any further requirements on anticipated transient without scram which may be imposed as a result of the rulemaking.

Based on our review, we conclude that there is reasonable assurance that the LRG-II plants can be operated prior to ultimate resolution of this generic issue without endangering 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 a nuclear reactor pressure vessel, three considerations are important. First, fracture toughness increases with increasing temperature; second, fracture toughness decreases with increasing load rates; and third, fracture toughness decreases with neutron irradiation.

In recognition of these considerations, power reactors are operated within restrictions imposed by the Technical Specifications on the 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 the 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.

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A.-11 Reactor Vessel Materials Toughness (Cont'd)

Based upon evaluation of the LRG-II reactor vessel's materials, toughness, we conclude that adequate safety margins exist for brittle failure during operating, testing, maintenance, and anticipated transient conditions over the life of the units. Since Task Action Plan A-11 is projected to be completed well in advance of LRG-II plants reactor vessels reaching a fluence level which would notably reduce fracture resistance, acceptable vessel integrity for the postulated accident conditions will be assured. When Task Action Plan 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.

The materials of the LRG-II reactor vessels meet the fracture toughness requirements of NB-2300 of the ASME Code. Based on this fact and the fabrication techniques employed on the vessel, we estimate that the total fluence over the design life would result in a final fracture toughness value above the minimum charpy impact requirement of 50 foot-pounds. In addition, the surveillance program required by Appendix H of 10CFR Part 50 will afford an opportunity to reevaluate the fracture toughness periodically during a minimum of the first half of the design life.

To assure adequate safety margins, adjustment to the nil ductility transient temperature (NDTT) and the development method for pressure/temperature curves are specified in 10CFR50 Appendices G and H. The amount of adjustment to the operating curves is a function of reference temperature, RT_{NDT} which depends upon the fast neutron (1 Mev) fluence and copper and phosphorus content in the RPV material. For BWR/6's, the copper and phosphorus content of the material is closely controlled. Furthermore, high upper shelf toughness is specified and all values for core belt line material were in excess of 75 ft-lbs. The fast neutron fluence is low with respect to other reactor types because of the additional moderator (water) in the annulus between the core shroud and the RPV. Therefore, the reactor pressure vessel material toughness (A-11) issue is of relatively low concern for BWR/6's.

Therefore, based upon the foregoing, we conclude that LRG-II plants can be operated prior to resolution of this generic issue without undue risk to the health and safety of the public.

A-17 Systems Interaction in Nuclear Power Plants

The licensing requirements and procedures used in the LRG-II plant safety reviews address many different types of systems interaction. Current licensing requirements are founded on the defense-in-depth principle. 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.

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A-17 Systems Interaction in Nuclear Power Plants (Cont'd)

These design provisions supplemented by the current review procedures of the Standard Review Plan, 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 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 organizational units and assign secondary responsibility to other units where there is a functional or interdisciplinary relationship. Designers follow somewhat similar procedures and provide for interdisciplinary reviews and analyses of systems. Task A-17 provide an independent study of methods that 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 I investigation was structured to identify areas where interactions are possible between and among systems and have the potential for negating or seriously degrading the performance of safety functions. The study concentrated on common cause of 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 I 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 are not within the Phase I scope of the Sandia Study.

The Systems Interaction Branch, formed in the Office of Nuclear Reactor Regulation 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 contracts, is underway; a range of methods is being considered and tested for feasibility againt a sample of some systems interaction candidates derived from Licensee Event Report evaluations.

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A-17 Systems Interaction in Nuclear Power Plants (Cont'd)

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.

LRG-II participants will provide for a systematic visual inspection by a multidisciplinary team to review the "as-built" condition of the plant areas where physical interactions could potentially result in adverse effects on safety-grade equipment. Visual inspections of interaction areas are performed for spatially coupled systems interactions initiated by seismic events. Any spatial separations that do not meet established design criteria are reported for disposition by analysis and/or hardware modification. LRG-II participants are improving their programs based on the experience gained in the industry's efforts, but will maintain the multidisciplinary team concept which the staff considers essential to a systems interaction analysis.

Therefore, we conclude that these is reasonable assurance that the LRG-II plants can be operated prior to final resolution of this generic issue without endangering the health and safety of the public.

A-39 Safety Relief Valve Hydrodynamic Loads

All BWR plants are equipped with a number of SRVs to control primary system pressure transients. The SRVs are mounted on the main steam lines inside the drywell with discharge lines routed through the drywell into the suppression pool. When an SRV is actuated, the steam released from the primary system is discharged into the suppression pool where it is condensed.

Actuation of an SRV can be either automatic, at a preset pressure, or manual by means of an external signal. A preselected number of SRVs are used for the Automatic Depressurization System (ADS) which is designed to reduce the reactor pressure and permit operation of the low pressure emergency core cooling systems. The ADS performs this function by automatic actuation of the specified SRVs following receipt of specific signals from the reactor protection system.

Upon actuation of an SRV, the air column within the partially submerged discharge line is compressed by the high pressure steam and, in turn, accelerates the water leg into the suppression pool. The water jets thus formed create pressure and velocity transients which are manifested as drag or jet impingement loads on submerged structures.

A-39 Safety Relief Valve Hydrodynamic Loads (Cont'd)

Following water clearing, the compressed air is also accelerated into the suppression pool, forming high pressure air bubbles. These bubbles execute a number of oscillatory expansions and contractions while rising to the suppression pool surface. The associated transients again create drag loads on submerged structures as well as pressure loads on the submerged boundaries. These loads are referred to as SRV air clearing loads. Containment structures, equipment and piping at LRG-II plants have been designed to accommodate these loads.

In July, 1976, the staff issued acceptance criteria for SRV loads for the Mark III containments. These criteria were established on the basis of our evaluation of the methodology for predicting the SRV loads which was proposed by the General Electric Company. In late 1980, however, GE proposed a revised method, which will result in substantial reduction of SRV loads. This improved method was based on the Caorso inplant SRV tests which were performed in January, 1979, in Italy. NRC has approved the revised GE method in NUREG-0802. The LRG-II plants have used the revised SRV loads accepted by the NRC and will review the inplant testing results from Kuo Sheng 1 and Grand Gulf-1, to determine their applicability and confirm the conservatism of their designs. This concern is addressed in further detail in LRG-II issue 1-CSB.

A-40 Seismic Design Criteria - Short-Term Program

NRC regulations required that nuclear power plant 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, these 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, re-reviews 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 sections of the Standard Review Plan 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 the LRG-II plants have been established on 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 Clinton, Perry and River Bend will be reevaluated on a case by case basis.

A-40 Seismic Design Criteria - Short-Term Program (Cont'd)

Accordingly, we have concluded that the LRG-II plants 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. Loss of the ability to draw water from the suppression pool could disable the emergency core cooling system.

The concern addressed by this Task Action Plan for boiling water reactors is primarily focused on the potential for degraded emergency core cooling system 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 III containment due to the large depth of the pool and the low approach velocities. LRG-II plants have a minimum suction submergence for the ECCS systems of over 7 feet. This concern is addressed in further detail in LRG-II issue 7-RSB.

With regard to potential blockage of the intake lines, the likelihood of any insulation being drawn into an emergency core cooling system pump suction line is very small. The potential debris in the drywell could only be swept into the suppression pool via the horizontal vents. Any pieces reaching the pool would tend to settle on the bottom and would not be drawn into the pump suction since the suction center line is minimum of 4 feet and the approach velocity is only 1 foot/second above the pool bottom. In addition, boiling water reactor designs employ strainers on the suction sized with flow areas 200 percent larger than the suction piping.

Accordingly, we conclude that LRG-II plants 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 two offsite alternating current power connections, a standby emergency diesel generator alternating current power supply, and direct current sources.

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A-44 Station Blackout

Task A-44 involves a study of whether or not nuclear power plants should be designed to accommodate a complete loss of all alternating current power, i.e., a loss of both the offsite and the emergency diesel generator alternating current power supplies. This issue arose because of operating experience regarding the reliability of alternating current 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 event, the onsite emergency alternating current power supplies were available to supply the power needed by vital safety equipment. However, in some instances, one of the redundant emergency power supplies have 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 alternating current power was not a design basis event for the LRG-II facilities. Nonetheless, a combination of design, operating, and testing requirements have been imposed to assure that these units will have substantial resistance to a loss of all alternating current and that, even if a loss of all alternating current should occur, there is reasonable assurance the core will be cooled. These design, operating, and testing requirements are discussed below.

A loss of offsite alternating current 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 SER.

If offsite alternating current 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 Sections 8.3 and 9.5 of the SER. The requirements include preoperational testing to assure the reliability of the installed diesel-generators.

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

The issue of station blackout was considered by the Atomic Safety and Licensing Appeal Board (ALAB-603) for the St. Lucie 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 accommodate a station blackout event. The Commission has reviewed their recommendations and determined that some interim measures should be taken at all facilities including LRG-II plants while Task A-44

A-44 Station Blackout (Cont'd)

is being conducted. NRC Generic Letter 81-04 requested a review of plant capability to mitigate a station blackout event and prompt implementation, as necessary, of emergency procedures and a training program for station blackout events. Consequently, interim emergency procedures and operator training for safe operation of the facility and restoration of alternating current power will be implemented. This action will be completed by fuel load date.

Based on the above, we have concluded that there is reasonable assurance that LRG-II plants 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 product 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 is provided to maintain primary system inventory, if alternating current power is not available. When the system is at low pressure, the decay heat is removed by the residual heat removal systems. This "Unresolved Safety Issue" 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 from 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 LRG-II plants are designed with various methods for the removal of decay heat. As discussed above, the decay heat is normally rejected to the turbine condenser and condensate is returned to the vessel by the feedwater system. The reactor core isolation cooling (RCIC) system provides an alternate means of supplying makeup water to the vessel. This turbine driven pump takes suction from the condensate storage tank and pumps to the vessel. 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 reactor core isolation cooling system is not available. Both of these systems (RCIC and HPCS) can supply water to the vessel from either the condensate storage tank or the suppression pool.

A-45 Shutdown Decay Heat Removal Requirements (Cont'd)

If the reactor core isolation cooling and high pressure core spray are unavailable, the reactor system pressure can be reduced by the automatic depressurization system so that cooling by the residual heat removal system can be initiated. When the condenser is not used, the heat rejected to the suppression pool is subsequently removed by the residual heat removal system.

The reactor core isolation cooling and high pressure core spray systems for the LRG-II plants have improvements over comparable systems at older boiling water reactors. The reactor core isolation cooling system has been upgraded to safety-grade quality (now required for all boiling water reactors), and the high pressure core spray is powered by its own dedicated diesel so it can operate with an assumed loss of all other sources of alternating current power. Also, the residual heat removal system contains three pumps; the flow capacity of any single pump (A or B) is sufficient to remove the decay heat.

Following the TMI accident, the industry performed and documented extensive analyses of feedwater transients and small-break loss-of-coolant accidents to support acceptability of current designs. A report of these analyses was provided to the NRC in NEDO-24708A Revision 1, dated December, 1980.

Based on the above, we have concluded that the LRG-II plants can be operated prior to the ultimate resolution of the 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 shutdown the plant, as well as equipment whose functions is not required for safe shutdown, but whose failure could result in adverse conditions which might impair shutdown functions.

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A-46 Seismic Qualification of Equipment in Operating Plants (Cont'd)

LRG-II plants were designed using current seismic design criteria, and methods for seismic equipment qualification are to be latest codes and standards. Requirements for seismic equipment qualification include IEEE 344-1975 and Regulatory Guides 1.92 and 1.100. Standard Review Plans 3.2.2, 3.9.2, 3.9.3, and 3.10 have also been considered in the qualification efforts.

Since identification of hydrodynamic load effects on LRG-II plant structures, an effort was initiated to assess the effects of these loads (in combination with previously established seismic loads) on equipment required to safely shut down the plant. This reassessment involved validation of equipment qualification through both analytical methods and additional testing, where required.

It is concluded that LRG-II plants 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 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 sever than those transients analyzed as anticipated operational occures. 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 rigorously 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 "Unresolved Safety Issue" is to define generic criteria that will be used for plant-specific reviews.

The LRG-II plant 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

A-47 Safety Implications of Control Systems (Cont'd)

independence between safety and nonsafety systems or providing isolating devices between safety and nonsafety systems. These devices preclude the propagation of nonsafety 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 "Unresolved Safety Issue," 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 system action. Specifically, these reviews include identification and evaluation of the potential adverse impacts to plant safety as a result of control system failures, effects from loss of non-Class 1E power sources, and harsh environments following high energy line breaks. These concerns are addressed in further detail in LRG-II issues 5-ICSB, 6-ICSB, and 7-ICSB.

A specific subtask of this "Unresolved Safety Issue" will be to study the reactor overfill transient in boiling water reactors to determine the need for preventative and/or mitigating design measures to preclude or minimize the consequences of this transient. Several early boiling water reactors have experienced reactor vessel overfill transients with subsequent two-phase or liquid flow through the safety/relief valves. Following these early events, commercial-grade high-level trips (Level 8) have been installed at LRG-II plants 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. In addition BWR/6's have a high level scram that precludes this concern.

Based on the above, we have concluded that there is reasonable assurance that LRG-II plants 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 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

A-48 Hydrogen Control Measures and Effects of Hydrogen Burns on Safety Equipment (Cont'd)

of certain construction materials by the spray solution; and (4) any synergistic chemical, thermal, and radiolytic effects of post-accidents 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, 10CFR Section 50.44, "Standards for Combustible Gas Control System in Light Water Cooled Power Reactors," and Criterion 41 of the General Design Criteria, "Containment Atmosphere Cleanup," in Appendix A to 10CFR 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.

Regulation 10CFR 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 10CFR 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 10CFR 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 small, 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 was developed and implemented. These interim measures were described in a second October 2, 1980, Federal Register notice.

For plants with Mark III containments, the proposed interim rule specified that either it must be demonstrated that the containment can withstand hydrogen burns or explosions or a detailed evaluation of possible hydrogen control measures must be performed and the selected measures installed.

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The LRG-II position is to comply with this interim rule through use of a hydrogen igniter system. This system consists of glow plug igniters distributed throughout the containment. This system is designed to ignite hydrogen at low concentrations, thereby maintaining the concentration of hydrogen below its detonable limit and preventing potential containment overpressure.

To collectively evaluate the concerns associated with the Hydrogen issue for Mark III containments, LRG-II participants are involved in an owners group. This group is sponsoring analytical work with General Electric, Offshore Power systems and others. Current evaluations of this group have demonstrated that containment pressures will remain well below the failure point as the result of the anticipated hydrogen release and burn.

Based on the above, we conclude that LRG-II plants can be operated prior to resolution of the "Unresolved Safety Issue" and the proposed rulemaking without undue risk to the health and safety of the public.

MJA:rf:csc/118A22

480.40 The response to questions 480.30 regarding a description of your program to improve the hydrogen control capability at Perry 1/2 is deficient. A program description needs to be provided in order for the staff to complete its review in a timely manner.

We need a description of 1) the system you propose to install; 2) the installation schedule; 3) its design bases; and 4) your research programs (including schedules) designed to demonstrate and/or confirm efficacy of the proposed system.

Response

The Cleveland Electric Illuminating Company as a participant in the Hydrogen Control Owner's Group (HCOG) is involved in a comprehensive program to improve the hydrogen control capability of Perry Nuclear Power Plant Units 1 and 2.

A Hydrogen Control Porgram Document was submitted on behalf of the HCOG in a letter from J. D. Richardson, HCOG Chairman to D. R. Denton dated January 15, 1982. This report identifies the tasks needed to satisfactorily address the use of igniter systems in Mark III containments. Both generic and plant specific tasks are included. A brief status of each of the tasks identified, with respect to Perry is attached.

A distributed igniter system has been selected for hydrogen mitigation at Perry. The design for this Hydrogen Control System (HCS) will be based on the igniter systems developed at Grand Gulf, Sequoyah, McGuire, and D. C. Cook Nuclear Stations. The system will be installed and operational prior to start-up. Proposed system design and specific locations are being evaluated and will be provided in a preliminary design report scheduled for submittal before April 30, 1982.

ATTACHMENT TO 480.40

	Task	Status
1.	Select Scenario	Scenario selection and development of hydrogen release rates have been completed by the HCOG with GE. The GE report evaluating the most probable accident scenario and providing generic hydrogen release rates will be submitted in early April. This report will be applicable to Perry.
2.	Select Mitigation System	Generic selection criteria were developed by HCOG, based on initial studies by MP & L for Grand Gulf Nuclear Station and the igniter mitigation system selected. Plant specific design features are being evaluated to verify the use of igniters at Perry.
3.	Design Hydrogen Ignition System	Specific igniter designs and generic design criteria for the hydrogen control system were developed through HCOG. PNPP is preparing a preliminary design report to provide details on the igniters, proposed locations, and system operation. This report is scheduled for submittal on April 30, 1982.
4.	Containment Ultimate Capacity Analysis	CEI submitted an interim report entitled "Ultimate Structural Capacity of Mark III Containment" in a letter from D. R. Davidson to R. L. Tedesco dated January 25, 1982.
5.	Selection of Containment Response Analysis Code	HCOG has completed this task with the development and the selection of CLASIX-3. A report entitled "CLASIX-3 Containment Response Sensitivity Analysis" was submitted by J. D. Richardson on behalf of HCOG, in a letter to H. R. Denton dated January 15, 1982.

Task	Status
5. Selection of Containment Response Analysis Code (Con't)	The analysis provides studies of the temperature and pressure response of a Mark III containment to hydrogen burns resulting from operation of an igniter system. This report is applicable to the Perry Nuclear Power Plant.
6. Containment Response Analysis	The HCOG has completed the generic Mark III Clasix-3 analysis and the sensitivity studies were submitted under Task 5 above. PNPP is contracting OPS to perform a plant specific base case analysis and will provide a containment response report in mid-1982.
7. Hydrogen Combustion Testing and Analysis	The HCOG has monitored the industry research and analysis regarding hydrogen, including the results of the ice condenser plants efforts. Specific analysis and testing to address the hydrogen combustion in the Mark III containment is being considered by HCOG in conjunction with the EPRI testing program. PNPP will participate in HCOG generic research efforts and will provide this information as it becomes available.
8. Equipment Survivability Analysis	A generic list of essential equipment which must survive the hydrogen burn is being developed by the HCOG and GE. Analysis will be provided to assure that the equipment will be capable of surviving postulated hydrogen burns at Perry. If applicable, generic heat transfer models of equipment will be developed by HCOG and used in the evaluation.