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Docket Nos. STN 50-522
and STN 50-523

Mr. Frank Spangenberg
Assistant Project Manager - Nuclear
Northwest Energy Services Company
2820 Northup Way,
Post Office Box 1090
Kirkland, Washington 98033

Dear Mr. Spangenberg:

Subject: Request for Information on Unresolved Safety Issues - Skagit/Hanford Nuclear Project, Units 1 & 2

We plan to include a Generic Safety Issues Appendix in the next Skagit/Hanford Safety Evaluation Report Supplement. The appendix will address those generic unresolved and technically resolved issues applicable to Skagit/Hanford.

In order for the NRC staff to include the Appendix in the next supplement, you should provide the information requested in Enclosure 1 by February 23, 1982, to enable us to complete our review and issue the SER supplement on schedule.

We have also provided for your information a copy of the Generic Issues Branch SER contribution for a recent BWR plant, Clinton, as Enclosure 2.

The reporting and/or recordkeeping requirements contained in this letter affect fewer than ten respondents; therefore, OMB clearance is not required under P.L. 96-511.

Sincerely,

Original signed by
Robert L. Tedesco

Robert L. Tedesco, Assistant Director
for Licensing
Division of Licensing

Enclosures:
As stated

cc w/encl:
See next page

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SKAGIT

Mr. J. E. Mecca, Manager
Nuclear Licensing & Safety
Puget Sound Power Light Co.
Puget Power Building
Bellevue, Washington 98009

cc: Mr. F. Theodore Thomsen
Perkins, Coie, Stone, Olsen
& Williams
1900 Washington Building
Seattle, Washington 98101

Mr. Robert Lowenstein
Lowenstein, Newman, Reis
& Axelrad
Suite 1214
1025 Connecticut Avenue, N.W.
Washington, D. C. 20036

Roger M. Leed, Esq.
Law Offices
1411 4th Avenue
Seattle, Washington 98104

Mr. Lloyd K. Marbet
c/o Forelaws on Board
19142 South Bakers Ferry Road
Boring, Oregon 97009

Mr. Nicholas D. Lewis, Chairman
Energy Facility Site Evaluation
Council
820 East 5th Avenue
Olympia, Washington 98504

Honorable Richard Sandvik
Department of Justice
500 Pacific Building
520 Southwest Yamhill
Portland, Oregon 97204

Coalition for Safe Power
Governor Building - Suite 527
408 S.W. Second Avenue
Portland, Oregon 97204

Mr. Russell Jim
Tribal Councilman
Consolidated Tribes and Bands
Yakima Indian Nation
P.O. Box 151
Toppenish, Washington 98948

Robert Engelken, Regional Administrator
U.S. Nuclear Regulatory Commission,
Region V
1450 Maria Lane, Suite 210
Walnut Creek, California 94596

REQUEST FOR INFORMATION

The Atomic Safety and Licensing Appeal Board in ALAB-444 determined that the Safety Evaluation Report for each plant should contain an assessment of each significant unresolved generic safety question. It is the staff's view that the generic issues identified as "Unresolved Safety Issues (NUREG-0606) are the substantive safety issues referred to by the Appeal Board. Accordingly, we are requesting that you provide us with a summary description of your relevant investigative programs and the interim measures you have devised for dealing with these issues pending the completion of the investigation, and what alternative courses of action might be available should the program not produce the envisaged result.

There are currently a total of 26 Unresolved Safety Issues. We do not require information from you at this time for a number of the issues since a number of the issues do not apply to your type of reactor, or because a generic resolution has been issued. However, we do request the information noted above for each of the issues listed below:

1. Waterhammer (A-1)
2. Reactor Vessel Materials Toughness (A-11)
3. Systems Interaction in Nuclear Power Plants (A-17)
4. Seismic Design Criteria (A-40)
5. Containment Emergency Sump Reliability (A-43)
6. Station Blackout (A-44)
7. Shutdown Decay Heat Removal Requirements (A-45)
8. Seismic Qualification of Equipment in Operating Plants (A-46)
9. Safety Implications of Control Systems (A-47)
10. Hydrogen Control Measures and Effects of Hydrogen Burns on Safety Equipment (A-48)

In addition to information on the above Unresolved Safety Issues, we request that you provide us with information on how the following technically resolved safety issues will be incorporated into the design, construction and/or operation of the Skagit/Hanford Nuclear Project:

Task Number	NUREG Report and Title
A-9	NUREG-0460, Vol. 4, "Anticipated Transients Without Scram for Light Water Reactors"
A-10	NUREG-0619, "BWR Feedwater Nozzle and Control Rod Drive Return Line Nozzle Cracking"
A-24	NUREG-0588, Revision 1, "Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment"
A-31	SRP 5.4.7 and BTP 5-1, "Residual Heat Removal Systems" incorporate requirements of USI A-31.

Task Number	NUREG Report and Title
A-36	NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants"
A-39	NUREG-0802, "Safety/Relief Valve-Quencher Loads Evaluation Report BWR Mark II and III Containments"
A-42	NUREG-0313, Revision 1, "Technical Report on Material Selection and Processing Guidelines for BWR Coolant Pressure Boundary Piping"

APPENDIX C

NUCLEAR REGULATORY COMMISSION (NRC) UNRESOLVED SAFETY ISSUES

C.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 (ACRS) 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 issued 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" (NUREG-0410, "NRC Program for the Resolution of Generic Issues Related to Nuclear Power Plants," dated January 1, 1978). 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.

C.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 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 would 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 Virginia Electric and Power Company (North Anna Nuclear Power Station, Unit Nos. 1 and 2), ALAB-491, NRC 245 (1978).

C.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"

"SECTION 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 Fiscal Year 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 action 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, NUREG-0410, entitled "NRC Program for the Resolution of Generic Issues Related to Nuclear Power Plants," describing the NRC generic issues program. 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.

The review is described in a report, NUREG-0510, "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 a 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, seventeen "Unresolved Safety Issues" addressed by twenty-two 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. Waterhammer - (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 under continuing study which have... potentially significant public safety implications." Six of the twenty-two tasks identified with the "Unresolved Safety Issues" are not applicable to Clinton because they apply to pressurized water reactors only. These tasks are A-2, A-3, A-4, A-5, A-12, and A-26. Also, tasks A-6, A-7, and A-8 only apply to Mark I or Mark II boiling water reactor containments. With regard to the remaining 13 tasks that are applicable to Clinton the NRC staff has issued NUREG reports providing its resolution of seven of the issues. The table below lists those issues.

Task Number	NUREG Report and Title	SER/SER Suppl. Section(s)*
A-9	NUREG-0460, Vol. 4, "Anticipated Transients Without Scram for Light Water Reactors"	
A-10	NUREG-0619, "BWR Feedwater Nozzle and Control Rod Drive Return Line Nozzle Cracking"	
A-24	NUREG-0588, Revision 1, "Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment"	
A-31	SRP 5.4.7 and BTP 5-1 "Residual Heat Removal Systems" incorporate requirements of USI A-31.	
A-36	NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants"	
A-39	NUREG-0802,** "Safety/Relief Valve-Quencher Loads Evaluation Report BWR Mark II and III Containments"	
A-42	NUREG-0313, Revision 1 "Technical Report on Material Selection and Processing Guidelines for BWR Coolant Pressure Boundary Piping"	

*Not available at this time. To be provided by the Project Manager.
**This report is scheduled for publication before the end of 1981. The report documents NRC acceptance of the SRV loads proposed by General Electric in Report #22A7000 Rev. 1 Appendix 3B for GESSAR II 238 Nuclear Island dated 11/25/80.

The remaining issues applicable to Clinton are listed in the following table.

GENERIC TASKS ADDRESSING
"UNRESOLVED SAFETY ISSUES"
THAT ARE APPLICABLE TO CLINTON UNIT 1

1.	A-1	Waterhammer
2.	A-11	Reactor Vessel Materials Toughness
3.	A-17	Systems Interaction in Nuclear Power Plants
4.	A-40	Seismic Design Criteria
5.	A-43	Containment Emergency Sump Reliability
6.	A-44	Station Blackout

With the exception of Tasks 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." 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 tasks; and a description of potential problems that could alter the planned approach or schedule.

In addition to the Task Action Plans, the staff issues the "Aqua Book" (NUREG-0606) on a quarterly basis. This book entitled, "Unresolved Safety Issues Summary, Aqua Book," provides current schedule information for each of the "Unresolved Safety Issues." It also includes information relative to the implementation status of each "Unresolved Safety Issue" for which technical resolution is complete.

We have reviewed the six "Unresolved Safety Issues" listed above and the four new USIs discussed in Section C.4 as they relate to Clinton Unit 1. Discussion

of each of these issues including references to related discussions in the Safety Evaluation Report is provided below in Section C.5. We have satisfactorily concluded our review for all but A-46, "Seismic Qualification of Equipment in Operating Plants" and A-48, "Hydrogen Control Measures and Effects of Hydrogen Burns on Safety Equipment." We will discuss resolution of these issues in a supplement to this Safety Evaluation Report. Based on our review, we have concluded for the reasons set forth in Section C-5 that, with the exception of A-46 and A-48, there is reasonable assurance that Clinton 1 can be operated prior to the ultimate resolution of these generic issues without endangering the health and safety of the public.

C.4 New "Unresolved Safety Issues"

An in-depth and systematic review of generic safety concerns identified since January 1979 has been performed by the staff to determine if any of these issues should be designated as new "Unresolved Safety Issues." The candidate issues originated from concerns identified in NUREG-0660, "NRC Action Plan as a Result of the TMI-2 Accident," ACRS recommendations, abnormal occurrence reports, and other operating experience. The staff's proposed list was reviewed and commented on by the ACRS, the Office of Analysis and Evaluation of Operational Data (AEOD) and the Office of Policy Evaluation. The ACRS and AEOD also proposed that several additional "Unresolved Safety Issues" be considered by the Commission. The Commission considered the above information and approved the following four new "Unresolved Safety Issues":

A-45	Shutdown Decay Heat Removal Requirements
A-46	Seismic Qualification of Equipment in Operating Plants
A-47	Safety Implication of Control Systems
A-48	Hydrogen Control Measures and Effects of Hydrogen Burns on Safety Equipment

A description of the above process together with a list of the issues considered is presented in NUREG-0705, "Identification of New Unresolved Safety Issues Relating to Nuclear Power Plants, Special Report to Congress," dated March 1981. An expanded discussion of each of the new "Unresolved Safety Issues" is also contained in NUREG-0705.

The applicability and bases for licensing prior to ultimate resolution of the four new USIs for Clinton Unit 1 is discussed in Section C.5.

C.5 Discussions of Tasks as They Relate to Clinton Unit 1

This section provides the NRC staff's evaluation of Clinton Unit 1 for each of the applicable "Unresolved Safety Issues." This includes our bases for licensing prior to ultimate resolution of these issues. Our conclusions are based in part on information provided by the applicant in their letter of October 23, 1981 from G. E. Wuller, Illinois Power Company to R. L. Tedesco, NRC.

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 and boiling water reactors have been reported. The waterhammers (or steam hammers) have involved steam generator feedrings and piping, the 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 approximately 118 actual and probable events have been reported in boiling water reactors as of September 1979, none have caused major pipe failures in a boiling water reactor such as Clinton 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 Clinton Power Station emergency core cooling system against the effects of waterhammer, the ECC systems are provided with jockey pumps which provide a continuous supply of water to the emergency core cooling system discharge piping. 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 a Technical Specification requirement to periodically vent air from the lines has been imposed. 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 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, Illinois Power 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 and 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 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 which have not explicitly been accounted for in the design and operation of Clinton, 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 concude that the Clinton Power Station can be operated prior to ultimate resolution of this generic issue 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 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.

Based upon our evaluation of the Clinton reactor vessel materials toughness, we conclude that adequate safety margins exist for brittle failure during operating, testing, maintenance, and anticipated transient conditions. 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 Clinton reactor vessel meet the fracture toughness requirements of NB-2300 of the ASME Code. Based on this fact and the fabrication techniques employed on the Clinton vessel, it is estimated that the total fluence over the design life would result in a final fracture toughness value above the minimum charpy impact requirement of 50 ft-lbs. In addition, the surveillance program required by Appendix H of 10 CFR 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 transition temperature (NDTT) and the development method for pressure/temperature curves are specified in 10 CFR 50 Appendices G and H. The amount of adjustment to the operating curves is a function of the fast Neutron (greater than 1 Mev) fluence and the copper and phosphorus content of 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.

In Clinton's case, the reactor pressure vessel (RPV) limiting material in the core belt line contains 0.10% copper and 0.016% phosphorus. The initial RT_{NDT} is -30°F. Based on a predicted adjusted reference temperature as a function of fluence and copper and phosphorous content, the end-of-life RT_{NDT} is predicted to be 41°F. On this basis, the Clinton RPV has adequate safety margin with respect to the requirements of 10 CFR 50 Appendices G and H.

Therefore, based upon the foregoing, we conclude that Clinton 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 staff's systems interaction program was initiated in May 1978 with the definition of USI A-17 (Systems Interaction in Nuclear Power Plants) and was intensified by TAP (NUREG-0660) Item II.C.3 (Systems Interaction). The concern arises because the design, analysis, and installation of systems are frequently the responsibility of teams of engineers with functional specialties--such as civil, electrical, mechanical, or nuclear. Experience at operating plants has led to questions of whether the work of these functional specialists is sufficiently integrated to enable them to minimize adverse interactions among systems. Some adverse events that occurred in the past might have been prevented if the teams had assured the necessary independence of safety systems under all conditions of operation.

The Illinois Power Company has not described a comprehensive program that separately evaluates all structures, systems, and components important to safety for the three categories of adverse systems interactions, i.e., spatially coupled, functionally coupled, and humanly coupled. However, there is assurance that Clinton-1 can be operated without endangering the health and safety of the public. The plant has been evaluated against current licensing requirements that 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 hazards such as high energy line ruptures, missiles, high winds, flooding, seismic events, fires, human factors, and sabotage. These design provisions are subject to review against the Standard Review Plan (NUREG-75/087) which requires interdisciplinary reviews of safety-grade equipment and address different types of potential systems interactions. Also, the quality assurance program which is followed during the design, construction and operational phases for each plant contributes to the prevention of introducing adverse systems interactions. Thus, the current licensing requirements and procedures provide an adequate degree of plant safety.

The NRC staff's current procedures assign primary responsibility for review of various technical areas to specific organizational units and assign secondary responsibility to other units where there is a functional interface. Designers follow somewhat similar procedures and provide the analyses of systems and interface reviews. Task A-17 has been developing methods that could identify adverse systems interactions which were 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 systems and have the potential of negating or seriously degrading the performance of safety functions. The study concentrated on commonly caused failures among systems that would violate a safety function. The investigation was to then identify where NRC review procedures may not have properly accounted for these interactions.

The Sandia Laboratories used fault-tree analysis on a selected LWR design to identify component failure combinations (cut-sets) that could result in loss of a safety function. The cut-sets were further reduced by incorporating six linking systems failures into the analysis. The results of the Sandia effort indicated a few potentially adverse systems interactions within the limited scope of the study. The staff reviewed the interactions for safety significance and generic implications. The staff concluded that no corrective measures needed to be implemented immediately except for the potential interaction between the PORV and its block valve. This interaction had been

separately identified by the evaluations of the TMI-2 accident while Sandia was studying the selected LWR. Since corrective measures were already being implemented, no separate measures were needed under USI A-17.

The Illinois Power Company has taken some initial steps toward the performance of a separate evaluation of Clinton-1 from a multidisciplinary point of view. They have formed a Systems Interaction Survey Team to analyse the "as-built" plant in areas where spatially coupled systems interactions could adversely affect safety-grade inspection equipment.

The "NRC Action Plan Developed as a Result of the TMI-2 Accident," NUREG-0660, provides for a systems interaction follow-on study, Section II.C.3, "Systems Interactions." Since April 1980, the Office of Nuclear Reactor Regulation has intensified the effort both by broadening the study of methods to identify potential systems interactions and by preparing guidance for audit reviews of selected plants for systems interactions. Our recent experience provides a basis from which we are developing a more efficient review process for potential systems interactions. The process will provide for a resolution of USI A-17, assimilate operating reactor experience, and rank identified systems interactions by their relative importance to safety.

It is expected that the development of systematic ways to identify, rank, and evaluate systems interactions will go further to reduce the likelihood of intersystem failures resulting in the loss of plant safety functions. A comprehensive program is expected to employ analytical methods, visual inspections, experience feedback, and simulator dependencies experiments. The LWR industry's current experience with systems interaction reviews is fragmented. Experience like that gained by the Phase I study is an essential ingredient to the staff's considerations of a comprehensive systems interaction program. After the resolution of USI A-17, we will determine whether Illinois Power must perform further evaluations for adverse systems interaction.

A-40 Seismic Design Criteria - Short-Term Program

NRC regulations require 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, 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, 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 Clinton have been established on existing licensing criteria and requirements. The staff's review of Clinton to these criteria is discussed in Section ____ of this Safety Evaluation Report. Should the resolution of Task A-40 indicate a change is needed in these licensing requirements, all operating reactors including Clinton will be reevaluated on a case by case basis.

Accordingly, we have concluded that Clinton 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 cooling and containment spray systems.

The concern addressed by this Task Action Plan for boiling water reactors is limited to 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 depth of the ECCS suction lines and the low approach velocities. The minimum ECCS suction submergence is 10 feet.

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 8 feet above the pool bottom. In addition, boiling water reactor designs employ strainers on the suction sized with flow areas 200% larger than the suction piping.

Accordingly, we conclude that Clinton 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.

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 events, 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 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 alternating current power was not a design basis event for the Clinton facility. Nonetheless, a combination of design, operating, and testing requirements have been imposed to assure that this unit 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 9.6 of the SER. Our 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 (March, 1983), 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 their recommendations and determined that some interim measures should be taken at all facilities including Clinton while Task A-44 is being conducted. 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 Clinton 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 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 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 Clinton reactor has 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 system provides an alternate means of supplying makeup water to the vessel. This turbine-driven pump takes suction from the RCIC 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 system is provided if the reactor core isolation cooling system is not available.

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 at Clinton 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 is sufficient to easily 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. The staff's assessment of current designs related to loss-of-feedwater transients and small loss-of-coolant accidents is contained in NUREG-0626, "Generic Evaluation of Feedwater Transients and Small Break Loss-of-Coolant Accidents in GE-Designed Operating Plants and Near-Term Operating License Applications."

Based on the above, we have concluded that Clinton can be operated prior to the 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.

Clinton was designed using current seismic design criteria, and commitments for seismic equipment qualification are to the latest codes and standards. Requirements for seismic equipment qualification in the Clinton FSAR include IEEE 344-1975 and Regulatory Guides 1.92 and 1.100. Standard Review Plans 3.9.2 and 3.10 have also been considered in the qualification efforts.

Since identification of hydrodynamic load effects on Clinton structures, an effort has been 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 includes equipment not required for safe shutdown, but whose failure could adversely affect equipment required for safe shutdown.

This reassessment involves validation of equipment qualification through both analytical methods and additional testing, when required. It is anticipated that this effort will be completed by December 1981, and results will be reflected in an FSAR amendment to modify appropriate portions of Sections 3.9 and 3.10. The staff's evaluation of the applicants' Seismic Qualification Program will be provided in a supplement to this Safety Evaluation Report.

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 could 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 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 all of these concerns; is is possible to develop generic criteria that can be used for future plant-specific reviews. The purpose of this "Unresolved Safety Issue" is to verify the adequacy of existing criteria for control systems or propose additional generic criteria (if necessary) that will be used for plant-specific review.

The Clinton 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 grade system equipment faults such that operation of the safety-grade system equipment is not impaired.

A wide range of bounding transients and accidents is presently analyzed to asure 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 the control system failures (single or multiple) will not defeat safety system action.

Also, the applicant has been requested (NRC Information Notice 79-22, "Qualification of Control Systems," September 17, 1979) to review the possibility of consequential control system failures which exacerbate the effects of high energy line breaks (HELB) and adopt new operator procedures where needed, to assure that the postulated events would be adequately mitigated. As part of the review, the staff is also evaluating the qualification program to assure that equipment that may potentially be exposed to HELB environments has been adequately qualified or an adequate basis has been provided for not qualifying the equipment to the limiting hostile environment. The staff's evaluation of the applicant's response to Information Notice 79-22 and the adequacy of the qualification program will be reported in Sections ___ and ___ of this Safety Evaluation Report, respectively.

With the recent emphasis on the availability of post-accident instrumentation (Reg. Guide 1.97, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant Conditions During and Following an Accident"), the staff reviews evaluate the designs to assure that control system failures will not deprive the operator of information required to maintain the plant in a safe shutdown condition after any "anticipated operational occurrence or accident." The applicant was requested to evaluate their control systems and identify any control systems whose malfunction could impact plant safety. The applicant is requested to document the degree of interdependence of these identified control systems and identify the use (if any) of common power supplies, and the use of common sensors or common sensor impulse lines whose failure could have potential safety significance. The results of these reviews and the staff's evaluation are documented in the Section ___ of the Safety Evaluation Report.

In addition, IE Bulletin 79-27 ("Loss of Non-Class IE Instrumentation and Control Power System Bus During Operation," November 30, 1979) was issued to the applicant requesting that evaluations be performed to ensure the adequacy of plant procedures for accomplishing shutdown upon loss of power to any electrical bus supplying power for instruments and controls. The results of this review are documented in the Safety Evaluation Report, Section ___.

The subtask of this issue concerning the reactor overfill transient in boiling water reactors is currently under review by the BWR Owners' Group of which Clinton is a member. Pending ultimate resolution of this item, the applicant has incorporated in the Clinton design, a commercial grade high level trip (Level 8) of the RCIC, HPCS, and feedwater systems to prevent the occurrence of overfill transients.

Based on the above, we have concluded that there is reasonable assurance that Clinton can be constructed and 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 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 Criterion 41 of the General Design Criteria, "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.

Regulation 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 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 such as Clinton, 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.

The Clinton Power Station will comply with this interim rule through use of a Hydrogen Ignition System (HIS). This system consists of glow plug igniters distributed throughout the containment and drywell. This HIS 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, an owners group has been formed. This group is sponsoring analytical work with General Electric, Offshore Power Systems and others. Current evaluations of this group indicate that containment pressures will remain well below the failure point as the result of the postulated hydrogen release and burn.

The staff is currently reviewing (1) the Hydrogen Ignition System, and (2) the applicant's analysis of the ability of essential equipment to survive the hydrogen burn environment. The results of this review will be provided in a supplement to this Safety Evaluation Report.