AUG 1 0 1981

Docket Nos: 50-390/391

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AUG 17 1981
U.S. NUCLEAR REGULATORY COMMISSION

Mr. H. G. Parris
Manager of Power
Tennessee Valley Authority
500 A Chestnut Street Tower II
Chattanooga, Tennessee 37401

Dear Mr. Parris:

SUBJECT: REQUEST FOR ADDITIONAL INFORMATION CONCERNING THE WATTS BAR NUCLEAR PLANT, UNITS 1 AND 2

Attached are requests for additional information developed as a result of our review of the Final Safety Analysis Report for the Watts Bar Nuclear Plant, Units 1 and 2. To expedite the review of your facility, these were forwarded to your staff informally on or prior to July 22, 1981.

Below is a list of the subject areas with submittal dates required to be met in order for us to meet our review schedule:

ENCLOSURE	Q. NOS.	SUBJECT	SUBMITTAL DATE
1 with attachment	371.28	Use of a permanent dewatering system	August 14, 1981
2	362.37-362.43	Geoscience Branch review of FSAR Chapter 2.5	August 14, 1981
3 with attachment	281.1 - 281.4	Chemical Engineering Branch review questions	August 10, 1981
4	450.1 - 450.2	Accident Evaluation Branch review questions	August 14, 1981
5 with attachment	N/A	Long-term operability of deep draft pumps	October 1, 1981
6 with attachment	N/A	Status of Unresolved Safety Issues	August 17, 1981
7	362.44	Bollinger report regarding Seismic activity in the Giles County area	August 17, 1981

APP3

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SURNAME	PDR ADOCK 05000390						
DATE	A PDR						

NRC FORM 318 (10-80) NRCM 0240 OFFICIAL RECORD COPY USGPO: 1981—3

Upon receipt of the information regarding the generic Unresolved Safety Issues, we may wish to set up a meeting with your staff to discuss the submittal.

If you have any questions concerning these matters, or can not meet the above submittal dates, please contact the project manager, T. J. Kenyon.

Sincerely,

/S/

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

Enclosures: As stated

cc: See next page

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WATTS BAR

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ENCLOSURE (1)

REQUEST FOR ADDITIONAL INFORMATION

371.28 In response to Question No. 371.23 (Am. 38), you stated that use of a permanent dewatering system is required to permanently lower groundwater levels at safety-related structures. This was the first indication of the use of such facilities.

In order to complete our review, we require additional information. The information needs and staff requirements for dewatering systems are found in Branch Technical Position (BTP) HMS/GSB 1, attached to Section 2.4.13 of the Standard Review Plan. A copy of this BTP has been provided to TVA informally, and is provided as an attachment to this request.

ATTACHMENT TO ENCLOSURE (1)

BRANCH TECHNICAL POSITIONS HMB/GSB 1

SAFETY-RELATED PERMANENT DEWATERING SYSTEMS

I. Summary

This position has been formulated to minimize review problems common to permanent dewatering systems that are depended upon to serve safety-related purposes by describing acceptable geotechnical and hydrologic engineering design bases and criteria. A safety-related designation for permanent dewatering systems is provided since they protect other safety-related structures, systems and components from the effects of natural and man caused events such as groundwater. In addition, the level of documentation of data and studies which are considered necessary to support safety-related functions is defined. This position applies to both active (e.g., uses pumps) and passive (e.g., uses gravity drains) dewatering systems. This position does not reflect structural, mechanical and electrical criteria.

II. Background

The staff has reviewed a number of permanent dewatering systems, including McGuire 1 & 2, Cherokee 1 & 2, Perkins 1 & 2, Perry 1 & 2, WPPSS 3 & 5, Douglas Point 1 & 2, and Catawba 1 & 2. Perry, beginning in 1975, was the first plant reviewed with such systems, and was reviewed very late in the CP process. Only WPPSS 3 & 5 and Douglas Point use a passive system (no pumps).

Permanent dewatering systems lower groundwater levels to reduce subsurface water loads on plant structures. In addition, they can increase plant operational dependability and reduce costs. These effects are accomplished by providing added means of keeping seepage water out of lower building levels during the later stages of plant life when normal waterproofing provisions may have deteriorated, and reducing radwaste system operating costs by minimizing the amount of drain water that must be treated. Benefits are, therefore, of two types, tangible (dollars) and intangible ("insurance"). We understand the construction costs of underdrains can vary widely depending on the design. Construction costs of between $125K to $1000K per unit have been suggested. The costs of coping with significant amounts of groundwater inleakage in safety-related building areas, which underdrains are expected to minimize, is estimated to be in the range of $100K to $200K per year per reactor. The construction costs of alternatives to underdrains for structural purposes alone (exclusive of inleakage treatment) is estimated to range upward from $300K per unit and is highly dependent on site conditions. Structural alternatives to permanent underdrains include additional concrete and steel in the lower portions of buildings, and the use of anchor systems to resist floatation.

Dewatering systems are generally composed of three components; the collector system, the drain system, and the discharge system. Water is first collected in collector drains adjacent to buildings or excavations. Interceptor drains or piping are then used to convey this water to a final discharge system. The discharge system can be either gravity flow or a pumping system. Most underdrain structures, systems and components are buried alongside and under structures, although some systems employ pumping systems within larger structures (such as reactor or auxiliary buildings) to discharge collected water. Finally, permanent dewatering systems are not a required feature at any plant, but may be proposed as a cost effective feature.

Many permanent dewatering systems at nonnuclear facilities, such as dams and large buildings, have functioned over the years. However, the likelihood of a portion of such a system becoming ineffective and, therefore, not performing its intended function may well be considerably greater than the probability of occurrence of a nuclear power plant design basis event such as a Probable Maximum Hurricane, Probable Maximum Flood, or Safe Shutdown Earthquake. Losses of function in the past have generally been attributable to piping of fines, inadequate capacity, or clogging. We have concluded that safety analyses of such systems should consider reliability and failures of features of the system itself, as well as potentially adverse effects of failures of nearby nonsafety-related features. Such systems need not be designed for design earthquakes if they are not intended to perform as underdrains fully during or immediately following a severe earthquake, or if the system can be expected to perform an underdrain function in a degraded condition. Certain portions of such systems, however, may be required to regularly perform other safety functions (e.g., porous concrete base mats) and should be designed for severe earthquakes. Failure of a dewatering system could cause groundwater levels to rise above design levels, resulting in overloading concrete walls and mats not designed to withstand the resulting hydrostatic pressures. In addition to causing potential structural and equipment damage, groundwater could enter safety-related buildings and flood components necessary for plant safety.

The basis for staff concerns over the use of such systems is whether they can be expected to perform their function, and prevent structural failures and interior flooding of safety-related structures. The degree of concern is directly related to the corresponding degree to which the safety of the structures and systems rely on the integrity of the dewatering system, particularly with a dewatering system in a degraded situation. For example, if structures can accommodate hydrostatic loads that would result with a total failure of a dewatering system, our concerns have been primarily limited to the capability of such systems to perform their functions under relatively infrequent earthquake situations. If, however, such systems must remain functional (e.g., keep water levels down), whether in a degraded situation or not to prevent structural failures and internal flooding under potentially frequent conditions, we have been very concerned with system reliability.

Many applicants have indicated that their plants can withstand, or have been designed against, full hydrostatic loadings that would occur in the absence of the underdrain systems, but not if an earthquake were to occur. If the plant can withstand full hydrostatic loading, assuming degradation of the underdrain system, many of the staff's concerns may be eliminated from further consideration because of the time available for remedial action after detection of system degradation.

III. Situations Identified During Previous Reviews

Four general categories of situations have been identified during case reviews as follows:

(a) Estimating and Confirming Permeability Values

It is necessary to estimate the amount of water that will be collected so that system components such as strip drains, blanket drains, collector pipes, and pumps are adequately designed and sized. One of the most important and most difficult parameters to evaluate is the permeability of the soil and rock existing at a site. A permeability value could be affected significantly by conditions of concentrated flow along joints in fractured and weathered rocks, or within other aquifers affected by foundation excavation. In addition, geological and foundation conditions that were not detected in site explorations may affect flow conditions and cause the estimated permeability values and flow regimes to be substantially different from those assumed at the CP preliminary design stage. These conditions are often first detected during construction dewatering. Therefore, we have required a commitment to consider construction excavation and dewatering data in the final design of underdrain systems. (See situation (d) below.)

(b) Operational Monitoring Requirements

To guard against system malfunctions and to assure sufficient time is available for implementation of remedial measures before groundwater could rise to an unacceptable level, provisions must be made for early detection of system failures, and contingency measures for these failures must be well defined prior to plant operation. Since drain systems are usually buried and concealed and there may be no direct way of inspecting them, reliance must be placed on piezometers, observation wells, manholes, and monitoring of collected water to detect problems or malfunctioning of the system. The details of an operational monitoring program are necessary prior to construction of the underdrain to assure that each of the following will be provided: (a) an early detection alarm system during normal operating conditions; (b) regularly scheduled inspection and monitoring; and (c) competent evaluation of observations during both construction and operation. In addition, the bases for acceptable contingency measures suitable for coping with various possible hazards must be established at the CP stage.

(c) Pipe Breaks

A dewatering system might be overloaded by such conditions as leaks or breaks in either the circulating or service water systems. A leak through a pipe break may be a very small percentage of the total flow of the cooling water system, but large enough to exceed the hydraulic capacity of drains, pipes and pumps in the dewatering system. For example, a complete failure of circulating water system piping has been required in the design of the dewatering systems reviewed to date. This requirement was made to assure that such abnormal occurrences do not adversely affect the integrity of safety-related structures, systems, and components.

(d) Sequence of Review

Underdrain systems are usually one of the first items constructed and, after backfilling and construction of subsurface facilities, are then no longer visible for regular inspection. In most cases, these systems are initially designed based on rather limited information from preconstruction field activities, and are tailored specifically for the site and facilities. By necessity then, final review and approval by the staff of the design must rely in some part on information gathered during construction. Therefore, the review and approval can be accomplished in two ways: (1) design details of the permanent underdrain system, the operational monitoring program and plans for construction dewatering can be submitted in the PSAR, with only confirmation of the details required prior to actual construction; or (2) conceptual designs of the permanent underdrain system and the operational monitoring program and details of construction dewatering can be submitted in the PSAR with the more complete review and approval based on construction dewatering requiring review and approval prior to actual construction. Review and approval of unique designs as post-CP matters is based upon 10 CFR Part 50, Subsections 35(b) and 55(e)(1)(iii). To prevent extending the review schedule, the first procedure would be the most desirable, but the staff recognizes that the detail required may not always be available at the time the PSAR is submitted.

IV. Proposed Staff Position

We have reviewed and approved the design of a limited number of permanent dewatering systems. However, because of the importance of these systems to plant safety, we have always required that they be designed and used in a conservative manner. The following is a list of required design provisions which are consistent with requirements in recent CP reviews:

(a) if the dewatering system is relied upon for any safety-related function, the system must meet the appropriate criteria of Appendix A and Appendix B to 10 CFR Part 50. In addition, guidance for structural, mechanical and electrical design criteria is provided in related sections of the Standard Review Plan for Category I structures, systems and components. However, all portions of the system need not be designed to accommodate all design basis events, such as earthquakes and tornados, provided that such events cannot either influence the system, or that the consequences of failure from such events is not important to safety; nevertheless, a clear demonstration of the effectiveness of a backup system and the timeliness of its implementation must be provided;

(b) the potential for localized pressures developing in areas which are not in contact with the drainage system, or in areas where pipes enter or exit the structural walls or mat foundations, must be considered.

(c) uncertainty in detecting operational problems and providing a suitable monitoring system must be considered;

(d) the potential for piping fines and clogging of filter and drainage layers must be considered;

(e) assurance must be provided that the system as proposed can be expected to reliably perform its function during the lifetime of the plant; and

(f) where the system is safety-related, is not totally redundant or is not designed for all design basis events, provide the bases for a technical specification to assure that in the event of system failure, necessary remedial action can be implemented before design basis conditions are exceeded.

V. SAR's (Std. Format & Content Information, Sections 2.4 & 2.b) for each of the plants with permanent dewatering systems should include the following information:

(a) Provide a description of the proposed dewatering system, including drawings showing the proposed locations of affected structures, components and features of the system. Provide information related to the geotechnical and hydrologic design of all system components such as interceptors, drainage blankets, and pervious fills with descriptions of material source, gradation limits, material properties, special construction features, and placement and quality control measures. (Note structural, mechanical and electrical information needs described elsewhere.) Where the dewatering system is important to safety, provide a discussion of its expected functional reliability. The discussion of the bases for reliability should include comparisons of proposed systems and components with the performance of existing and comparable systems and components for applications under site conditions similar to those proposed. Where such information is unavailable or unfavorable, or the application (design and/or site) is unique, the unusual features of the design should be supported by additional tests and analyses to demonstrate the conservative nature of the design. In such cases the staff will meet with the applicant, on request, to establish the bases for such additional tests and analyses.

(b) Provide estimates, and their bases, for soil and rock permeabilities, total porosity, effective porosity (specific yield), storage coefficient and other related parameters used in the design of the dewatering system. In general, these site parameters should be determined utilizing field and, if necessary, laboratory tests of materials representative of the entire area of influence of the expected drawdown of the system. Unless it can be substantiated that aquifer materials are essentially homogeneous, or that obviously conservative estimates have been used as design bases, provide preconstruction pumping tests and other in-situ tests performed to estimate the pertinent hydrologic parameters of the aquifer. Monitoring of pumping rates and flow patterns during dewatering for the construction excavation is also necessary to verify assumed design bases relating to such factors as permeability and aquifer continuity. In addition, the final design of the system should be based on construction dewatering data and related observations to assure that the values estimated from site exploration data are conservative. Lastly, the final design of the dewatering system and its hydrologic and geotechnical operational monitoring program should be confirmed by construction excavation and dewatering information.

If such information fails to support the conservatism of design information previously reviewed by the staff, the changed information should be reviewed under 10 CFR Part 50, Subsections 35(b) and 55(e)(1)(iii).

(c) Provide analyses and their bases for estimates of groundwater flow rates in the various parts of the permanent dewatering system, the area of influence of drawdown, and the shapes of phreatic surfaces to be expected during operation of the system. The extent of influence of the drawdown may be especially important if a natural or man-made water body affects, or is affected by, the dewatering systems.

(d) Provide analyses, including their bases, to establish conservative estimates of the time available to mitigate the consequences of system degradation* that could cause groundwater levels to exceed design bases. Document the measures that will be taken to either repair the system, or provide an alternate dewatering system that would become operational before the design basis groundwater level is exceeded.

(e) Provide both the design basis and normal operation groundwater levels for safety-related structures, systems and components. The design basis groundwater level is defined as the maximum groundwater level used in the design analysis for dynamic or static loading conditions (whichever is being considered), and may be in excess of the elevation for which the underdrain system is designed for normal operation. This level should consider abnormal and rare events (such as an occurrence of the Safe Shutdown Earthquake (SSE), a failure of a circulating water system pipe, or a single failure within the system), which can cause failure or overloading of the permanent dewatering system.

(f) A single failure of a critical active feature or component must be postulated during any design basis event. Unless it can be documented that the potential consequences of the failure will not result in Regulatory Guides 1.26 and 1.29 dose guidelines being exceeded, either (1) document by pertinent analyses that groundwater level recovery times are sufficient to allow other forms of dewatering to be implemented before the design basis groundwater level is exceeded, discuss the measures to be implemented and equipment needed, and identify the amount of time required to accomplish each measure, or (2) design for all system components for all severe natural phenomena and events. For example, if the design basis groundwater level can be exceeded only as a result of a single nonseismically induced failure of any component or feature of the system, the staff may allow the design basis level of the dewatering system to be exceeded for a short period of time (say 2 or 3 days), provided that (1) effective alternate dewatering means can be implemented within this time period, or that (2) it can be shown that Regulatory Guides 1.26 and 1.29 guidelines will not be exceeded by groundwater induced impairments of safety-related structures, systems, or components.

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*See (f) for considerations of differing system types.

(g) Where appropriate, document the bases which assure the ability of the system to withstand various natural and accidental phenomena such as earthquakes, tornadoes, surges, floods, and a single failure of a component feature of the system (such as a failure of any cooling water pipes penetrating, or in close proximity to, the outside walls of safety-related buildings where the groundwater level is controlled by the system). An analysis of the consequences of pipe ruptures on the proposed underdrain system must be provided, and should include considerations of postulated breaks in the circulating system pipes at, in, or near the dewatering system building either independently of, or as a result of the SSE. Unless it can be documented that the potential consequences will not be serious enough to affect the safety of the plant to the extent that Regulatory Guides 1.26 and 1.29 guidelines could be exceeded, provide analyses to document that (1) water released from the pipe break cannot physically enter the dewatering system, or (2) if water enters the dewatering system, the system will not be overloaded by the increased flow such that the design basis groundwater level is subsequently exceeded.

(h) State the maximum groundwater level the plant structures can tolerate under various significant loading conditions in the absence of the underdrain system.

(i) Provide a description of the proposed groundwater level monitoring programs for dewatering during plant construction and for permanent dewatering during plant operation. Monitoring information requested includes (1) the general arrangement in plan and profile with approximate elevation of piezometers and observation wells to be installed, (2) intended zone(s) of placement, (3) type(s) of piezometer (closed or open system), (4) screens and filter gradation descriptions, (5) drawings showing typical installations showing limits of filter and seals, (6) observation schedules (initial and time intervals for subsequent readings), (7) plans for evaluation of recorded data, and (8) plans for alarm devices to assure sufficient time for initiation of corrective action. Provide a commitment to base the final design of the operational monitoring program on data gathered during the construction monitoring program (if construction experience shows the assumed operational program bases to be nonconservative or impractical). Changes to the operational program are to be documented in the FSAR.

(k) Provide information regarding the outlet flow monitoring program. The information required includes (1) the general location and type of flow measurement device(s), and (2) the observation plan and alarm procedure to identify unanticipated high or low flow in the system and the condition of the effluent.

(l) For OL reviews, but only if not previously reviewed by the staff, provide (1) substantiation of assumed design bases using information gathered during dewatering for construction excavation, and (2) all other details of the dewatering system design that implement design bases established during the CP review.

(m) For OL reviews, provide a Technical Specification for periods when the dewatering system may be exposed to sources of water not considered in the design. An example of such a situation would be the excavation of surface seal material for repair of piping such that the underdrain would be exposed to direct surface runoff. In addition, where the permanent dewatering system is safety related, is not completely redundant, or is not designed for all design basis events, provide the bases for a technical specification with action levels, the remedial work required and the estimated time that it will take to accomplish the work, the sources, types of equipment and manpower required and the availability of the above under potentially adverse conditions. [See Section V(f)].

ENCLOSURE (2)

REQUEST FOR ADDITIONAL INFORMATION

362.37 The line of section for the regional cross-section of Fig. 2.5-3 is indicated on a small inset map of Tennessee. It is not possible to locate this line on the state geologic map of Eastern Tennessee. Please be more specific and indicate where on the state geologic map this line of section may be found.

362.38 P. 2.5-17 Para. 2: This paragraph discusses the Mississippian Pennington Formation and refers the reader to Fig. 2.5-9. However, this figure does not include the Pennington or any Mississippian rocks. Is the reference incorrect; do you mean to refer to another figure?

362.39 P. 2.5-21, Para. 5 - Reference 83 (Milici) was "in press" at time of the FSAR wriitng. Please furnish correct references.

362.40 P. 2.5-22 Para. 3 - To update the FSAR, you should discuss COCORP reflection findings also in discussion of thin-skin tectonics. Update the references also for p. 2.5-64 para. 2.

362.41 P. 2.5-24 - Para. 1 - refers to faulting described in Para. 2.4.1. However, no such discussion can be found in that paragraph.

362.42 P. 2.5-29 Para 1. - It is stated that Swingle's cross-section is based on information that confirms the sole of the thrust to be at 9,000 ft? What information was used to "confirm" this?

362.43 P. 2.5-54 Para 2 - What evidence supports the statement that the faults are confined to the Conesauga Formation and do not intersect any other stratigraphic formation?

ENCLOSURE (3)

ADDITIONAL INFORMATION REQUIRED BY
CHEMICAL ENGINEERING BRANCH FROM
WATTS BAR NUCLEAR PLANT, UNIT NOS. 1 AND 2

281.1 (9.1.3) Describe the samples and instrument readings and their frequency of measurement that will be performed to monitor the Spent Fuel Pool (SFP) water purity and need for SFP cleanup system demineralizer resin and filter replacement. State the chemical and radiochemical limits to be used in monitoring the SFP water and initiating corrective action. Provide the basis for establishing these limits. Your response should consider variables such as: boron concentration, gross gamma and iodine activity, demineralizer and/or filter differential pressure, demineralizer decontamination factor, pH, and crud level.

281.2 (9.3.2) (a) It is our position that provisions should be made in the process sampling system to purge and drain sample streams back to the system or origin, or to an appropriate water treatment system, in accordance with acceptance criterion II.2.e in Standard Review Plan Section 9.3.2. Indicate what provisions are available in your process sampling system for meeting this position.

(b) It is our position that automatic isolation valves in the process sampling lines that originate within the containment should fail in the closed position in accordance with acceptance criterion II.2.f in Standard Review Plan Section 9.3.2. Verify that this position is met in the process sampling system.

(c) Provide piping and instrumentation diagrams for the process sampling system.

281.3 (9.3.4) The Sequoyah-Watts Bar NSSS/BOP Comparisons only compare the component parameters without discussing differences in system design. Provide a comparison of any differences in the system design (flow paths, controls and alarms) of the Chemical and Volume Control system (CVCS) between the Watts Bar Nuclear Plant and Sequoyah Nuclear Plant. Provide a safety evaluation of the Watts Bar's CVCS which addresses any differences in system design.

281.4 (TMI II.B.3) Provide information that satisfies the attached proposed license conditions for post-accident sampling.

ATTACHMENT TO ENCLOSURE (3)

SAFETY EVALUATION
BY THE OFFICE OF NUCLEAR REACTOR REGULATION
WATTS BAR NUCLEAR PLANT, UNITS 1 AND 2
TENNESSEE VALLEY AUTHORITY
DOCKET NOS. 50-390/391

NUREG-0737, II.B.3 - Post Accident Sampling Capability

REQUIREMENT

Provide a capability to obtain and quantitatively analyze reactor coolant and containment atmosphere samples, without radiation exposure to any individual exceeding 5 rem to the whole body or 75 rem to the extremities (GDC-19) during and following an accident in which there is core degradation. Materials to be analyzed and quantified include certain radionuclides that are indicators of severity of core damage (e.g., noble gases, iodines, cesiums and non volatile isotopes), hydrogen in the containment atmosphere and total dissolved gases or hydrogen, boron and chloride in reactor coolant samples in accordance with the requirements of NUREG-0737.

To satisfy the requirements, the application should (1) review and modify his sampling, chemical analysis and radionuclide determination capabilities as necessary to comply with NUREG-0737, II.B.3, (2) provide the staff with information pertaining to system design, analytical capabilities and procedures in sufficient detail to demonstrate that the requirements have been met.

EVALUATION AND FINDINGS

The applicant has committed to a post-accident sampling system that meets the requirements of NUREG-0737, Item II.B.3 in Amendment , but has not provided the technical information required by NUREG-0737 for our evaluation. Implementation of the requirement is not necessary prior to low power operation because only small quantities of radionuclide inventory will exist in the reactor coolant system and therefore will not affect the health and safety of the public. Prior to exceeding 5% power operation the applicant must demonstrate the capability to promptly obtain reactor coolant samples in the event of an accident in which there is core damage consistent with the conditions stated below.

1. Demonstrate compliance with all requirements of NUREG-0737, II.B.3, for sampling, chemical and radionuclide analysis capability, under accident conditions.

2. Provide sufficient shielding to meet the requirements of GDC-19, assuming Reg. Guide 1.4 source terms.

3. Commit to meet the sampling and analysis requirements of Reg. Guide 1.97, Rev. 2.

4. Verify that all electrically powered components associated with post accident sampling are capable of being supplied with power and operated, within thirty minutes of an accident in which there is core degradation, assuming loss of off site power.

5. Verify that valves which are not accessible for repair after an accident are environmentally qualified for the conditions in which they must operate.

6. Provide a procedure for relating radionuclide gaseous and ionic species to estimated core damage.

7. State the design or operational provisions to prevent high pressure carrier gas from entering the reactor coolant system from on line gas analysis equipment, if it is used.

8. Provide a method for verifying that reactor coolant dissolved oxygen is at < 0.1 ppm if reactor coolant chlorides are determiend to be > 0.15 ppm.

9. Provide information on (a) testing frequency and type of testing to ensure long term operability of the post accident sampling system and (b) operator training requirements for post-accident sampling.

In addition to the above licensing conditions the staff is conducting a generic review of accuracy and sensitivity for analytical procedures and on-line instrumentation to be used for post-accident analysis. We will require that the applicant submit data supporting the applicability of each selected analytical chemistry procedure or on-line instrument along with documentation demonstrating compliance with the licensing conditions four months prior to exceeding 5% power operation, but review and approval of these procedures will not be a condition for full power operation. In the event our generic review determines a specific procedure is unacceptable, we will require the applicant to make modifications as determined by our generic review.

ENCLOSURE (4)

REQUEST FOR ADDITIONAL INFORMATION

450.1 (15.6.2) In evaluating the radiological consequences of the failure of small lines carrying primary coolant outside the containment, provide the following:

a. size and type of all small lines carrying primary coolant outside containment (including CVCS letdown line);

b. mass of reactor coolant released during accident;

c. summary of primary system iodine activity during the accident and its effects on the calculated accident consequences;

d. iodine transport mechanism and release path from the leak point to the environment;

e. isolation valve closure time and leakage rate;

f. detailed and chronological description of primary system response, including system response time,operator action, valve closure times, etc.;

g. figure indicating primary system pressure and temperature as a function of time during an accident;

h. figure indicating leak rate from the failure of small lines as a function of time.

450.2 The meteorological measurements tower is located close enough to the cooling towers that the measurements may be obstructed during down valley airflow. Provide analysis that will show the extent of the cooling tower influence on meteorological measurements made at the meteorological tower. This information should include data collected at the tower before and after the cooling tower construction.

ENCLOSURE (5)

REQUEST FOR ADDITIONAL INFORMATION
REGARDING THE
LONG-TERM OPERABILITY OF DEEP DRAFT PUMPS

IE Bulletin 79-15, dated July 11, 1979, was issued to all licensees and holders of construction permits as a result of deep draft pump deficiencies that were identified at facilities both operating and under construction. In your response to the bulletin you identified deep draft pumps as being utilized at your facility. However, your response to the bulletin did not include enough information to demonstrate and assure the long term operability of these pumps.

Attached is a document entitled "Guidelines for Demonstration of Operability of Deep Draft Pumps." In accordance with the schedule given in the cover letter, we request you provide information on all the deep draft pumps identified in your bulletin response, which describes the extent to which your deep draft pump long term operability assurance program conforms to the various portions of these Guidelines. Emphasis should be placed on 1) the establishment of installation procedures that are followed each time these pumps are disassembled and reinstalled, and 2) the testing requirements and bearing wear criteria. The instrumentation called for in the Guidelines should not be considered a requirement.

These Guidelines establish an acceptable method of assuring long term operability of deep draft pumps. They do not necessarily constitute the only method for demonstrating long term operability. The staff will review the information you submit to determine whether your long term operability assurance program for deep draft pumps is in sufficient conformance with these Guidelines to assure long term operability. If not, the staff will determine whether you have established and utilized other methods and procedures, preferably with the assistance of the pump manufacturer, that also demonstrate and assure that these pumps will perform their intended function for the length of time required. We anticipate completing our review and resolving this issue by January 1, 1982, or prior to issuance of your operating license.

ATTACHMENT TO ENCLOSURE (5)

GUIDELINES FOR DEMONSTRATION OF OPERABILITY OF DEEP DRAFT PUMPS

DISCUSSION

I.E. Bulletin 79-15 dated July 1979, identified problems associated with deep-draft pumps found at operating facilities and near term operating licensee facilities. Deep draft pumps, which are also called "vertical turbine pumps," are usually 30 to 60 feet in length with impellers located in casing bowls at the lowest elevation of the pump. The motor (driver) is located at the highest pump elevation with the discharge nozzle just below the motor.

Bulletin 79-15 was initiated because several nuclear power plant facilities could not demonstrate operability of their pumps. The pumps were experiencing excessive vibration and bearing wear. The rapid bearing wear suggested that these pumps could not perform their required functions during or following an accident. As a result of the staff's initial review of the responses to IEB 79-15, several plants were identified as having potential problems with their deep draft pumps. These guidelines are provided for these plants so that the licensee or applicant involved may have a method acceptable to the staff for demonstrating the operability of deep-draft pumps.

DEEP DRAFT PUMP OPERATING CHARACTERISTICS

In order to better understand the operating characteristics of these pumps, a rotor dynamics analyses[1] was performed to ascertain the response of the pump rotor under steady state operation. The analyses considered journal bearing to shaft dynamic response at various eccentricities and fluid viscosities. The model for the analysis depicted a typical deep draft pump utilized by the nuclear industry. The analysis resulted in recommendations for improving the stability of the pump rotor from externally applied inputs and by self-generated inputs.

The conclusions which were derived from the analysis and staff evaluations of North Anna, Beaver Valley and Surry facilities with similar pumps include:

1.) Pumps with this type of configuration are prone to bearing whirl vibration problems due to the flexibility of the rotor and casing structure. This phenomenon is accentuated as journal bearing clearance becomes large. This phenomenon leads to bearing wear (Journal bearings).

[1] "Low Head Safety Injection Pump Rotor Dynamic Analyses", by Franklin Research Center, Report FC4982, dated May 1980.

2.) There may be natural frequencies associated with the pump assembly which occur near the operating speed of the pump. Pump operation will drive these frequencies and can cause bearing wear. The severity of this condition is dependent on bearing diametral clearance, rotor unbalance conditions and housing flexibility. As an example, if the wear in column journal bearings becomes sufficiently large (twice the original diametral clearance) so that these bearings are no longer active and the undamped critical frequency near the operating speed of the pump is allowed to expand, the additional uncontrolled bearing wear will occur. This wear can continue until the shaft rubs against the support structure of the bearing and can potentially sever the shaft.

3.) One acceptable method for correcting instabilities in the pump shaft is to utilize a journal bearing design which exhibits stable characteristics. One such design is the "Taper land bearing". This design is more stable than the plain journal bearing, is less susceptible to wear because of the taper and will cause the bearing to form a hydrodynamic film quickly during startup.

4.) Stiffening of the column sections of the pump is advantageous if there is a column frequency near the operating speed of the pump. The shifting of the column frequency to a higher level will eliminate any coupling between the pump operating speed and the column frequency.

5.) Flow inlet conditions to the pumps and sump designs can be important to pump operability. Certain installations have demonstrated flow characteristics which produced vortexing at the bellmouth of the pump. This vortexing is due to sump design or sump supply line entrance conditions. This condition can contribute to additional pump vibration and wear. Flow straightener devices, reduction of bellmouth diameters, and bottom clearance reductions have proven to be effective in eliminating this problem.

6.) This type of pump has exhibited operational problems due to design and installation deficiencies. The high flexibility of the shaft and column make this design rather forgiving when it comes to installation deficiencies such as misalignment between the shaft and column, low-precision coupling assemblies, and non-perpendicular mounting flanges. This fact however, can lead to excessive bearing wear without significant noticeable change in pump operating characteristics. To ensure proper pump operation, proper alignment should be established between all mating surfaces and measures should be emphasized which prevent column and shaft eccentricities. These measures can include optical alignment of the column segments, use of high precision couplings and use of accurate techniques to establish that the sump plumb line is perpendicular to the pump mounting flange.

The above findings and conclusions have contributed significantly to the development of these guidelines. The guidelines listed below are divided into installation and test areas. The subjects to be addressed in these areas are considered to be of prime importance when establishing a pump operability assurance program. The extent to which each of the two areas are implemented at a specific facility is dependent on specific symptoms which have been identified with these pumps while in operation and during service periods.

Implementing the measures outlined below, at North Anna 1 & 2 in total, has been shown to provide reasonable assurance that the pumps will be operable when required for their safety function. These guidelines are not intended to replace the requirements of Standard Review Plan 3.9.3, Regulatory Guide 1.68 or any other requirements presently enforced by the staff. Rather, the guidelines are to be used as supplementary material for establishing deep-draft pump operability.

GUIDELINES FOR OPERABILITY INSTALLATION

1.0 INSTALLATION PROCEDURES

Experience has shown that these pumps are prone to having operability problems as a result of poor installation procedures. The guidelines emphasize those areas of the installation procedure, which if implemented, could significantly improve the likelihood of an operable pump. The procedures utilized should be submitted to the staff for review.

1.1 PUMP INSTALLATION

a. Determine by measurement that all shaft segments are straight within tolerances specified by the manufacturer.

b. Determine by measurement or provide certification that all couplings (for shaft segments & pump to motor coupling) are of high precision as specified by the manufacturer.

c. Determine by measurement that all pump segment flanges are perpendicular to the centerline of the segment, that the segments are straight and that any mating surfaces are concentric to an established datum. Where journal bearing guides (SPIDERS) are used, establish concentricity between this assembly and its mating surface.

d. Align full pump casing assembly optically to assure maximum straightness and concentricity of the assembly. Any equivalent method is acceptable, as long as the procedure stresses column straightness and concentricity.

e. Assure pump to motor flange perpendicularity and that proper coupling installation is performed .

f. Assure that all mating surface bolting is properly attached and that manufacturer torquing sequences are adhered to.

1.2 SUMP INSTALLATION

a. Assure (where used) that sump/pump mating flange is perpendicular to the sump pump line.

b. Assure that sump design prevents fluid anomalies such as vortexing or turbulence near the intake to the pump bellmouth and that incoming piping is not so designed as to allow fluid conditions favorable to these anomalies (i.e., sharp bends in piping prior to entrance into sump).

c. Assure that interference does not exist between the sump and any pump appendage such as a seismic restraint.

2.0 Testing Requirements

The installation procedures are essential in establishing pump operability. In addition to careful installation, testing may be required which will verify proper operation of these pumps. After completion of the installation checks, licensees or applicants should evaluate the need for further testing and report the results of this evaluation together with the details of any test plans to the staff. Should tests be required, an acceptable test procedure should include the items listed below. The staff recognizes that the instrumentation and procedures outlined below may be difficult to implement at all facilities and, therefore, the staff is emphasizing good installation practices which lead to operable components. If tests demonstrating operability cannot encompass all the items listed below, then alternative procedures should be proposed for evaluation by the staff. The tests should emphasize measurement of pump dynamic characteristics and wear data at different stages of testing, culminating with an extrapolation of the data to the desired life goal for the pump.

2.1 Test Instrumentation

The following instrumentation should be incorporated into the test procedure aside from normal flow measurement, pressure and vibration instrumentation:

a.) X, Y proximity probes at three axial locations on the pump column, for measuring and recording radial positions of shaft with respect to the column.

b.) X, Y, accelerometers (at proximity probe locations) for measuring and recording radial accelerations of the column.

c.) Dynamic pressure transducers for measuring fluid pressure at the following locations:

1. Bottom of Column (suction)
2. Mid-Column
3. Top of Column.

d.) Shaft Rotational speed and dynamic variation instrument.

2.2 PRE-TEST DATA

With the pump disassembled, measure all journal bearing O.D.'s, bearing I.D.'s and calculate bearing diametral clearances. In addition with pumps fully assembled and using the proximity probes, obtain the "clearance circle" at each of the three axial stations by rolling the shaft section within the clearance volume of its bearings and in this way, establish proper operation of the probes.

3.1 PHASE 1 Testing (6 hours plus start-stop)[2]

This phase of testing should be comprised of 6 hours of testing (Break-in) followed by start-stop testing. Test conditions should simulate as nearly as possible normal and accident conditions. Parameters to be considered are flow, temperature, debris, and chemical composition of fluid being pumped. Static torque tests should be performed before and after the test (i.e. measure amount of torque required to turn shaft by hand). Data should be taken during the six hour test at 1/2 hour intervals. A total of 12 start-stop tests will be performed consisting of a start up from zero speed up to full-speed, 10-minute dwell at full-speed and a shutdown from full speed to zero speed, with recording of all instrumentation during full cycle of start-stop.

Upon completion of Phase 1 testing, the following data should be obtained and recorded:

1.) Obtain the "clearance circles" using the three sets of proximity probes.

[2]Tests at North Anna 1 & 2 and Manufacturers input indicates that 6 hours is an adequate time interval for bearing "break in" period.

2.) Measure and record the following dimensions for each bearing:

a.) Journal O.D.

b.) Bearing I.D.

c.) Bearing to Journal diametral clearance

d.) Establish Phase 1 test bearing wear.

THE ACCEPTANCE CRITERIA IS AS FOLLOWS:

1.) If wear is $>$ 5 mils for any bearing,[3] wear is unacceptable and test should be terminated.

2.) If wear is $<$ 5 mils for all bearings[3]

a.) Reassemble the pump

b.) Obtain "clearance circles"

c.) Reinstall pump in test loop.

2.4 Phase 2 Testing (48 hours)

Phase 2 testing is to be performed at full system pressure and temperature and fluid conditions simulating those expected during accident and normal operation. Before start and at completion of Phase 2 test, obtain measurement of static torque. Data should be recorded continuously during the start-up period,

[3] This acceptable wear value may be modified based on manufacturers recommendation.

and during the shutdown period. Data should also be recorded at 1-hour time intervals during the 48 hour test.

The following measurements should be made at the completion of Phase 2 of the test:

1.) Obtain the "clearance circles" using the three sets of proximity probes.

2.) Measure and record the following dimensions for each bearing:

   a.) Journal O.D.

   b.) Bearing I.D.

   c.) Bearing to Journal diametral clearance.

   d.) Establish accumulated bearing wear.

THE ACCEPTANCE CRITERIA IS AS FOLLOWS:

1.) If accumulated bearing wear on any bearing is >7 mils , wear is unacceptable[3] and test should be terminated.

2.) If accumulated wear on all bearings is <7 mils for all bearings[3]:

   a.) Reassemble pump

   b.) Obtain "clearance circles".

   c.) Reinstall pump in test loop.

5.) Phase 3 Testing (96 hours)

Phase 3 testing is to be performed at full system pressure and temperature and fluid conditions simulating those expected during accident and normal operation. The same procedures should be followed as in Phase 2 testing except that data may be taken with less frequency.

The same measurements should be taken at the completion of this phase as with the other phases with the following acceptance criteria:

1.) If accumulated bearing wear is $>$ 8 mils for any bearing,[3] wear is unacceptable and test should be terminated.

2.) If accumulated wear is $<$ 8 mils for all bearings,[3] a decision needs to be made to establish:

a.) the need for additional testing or

b.) whether or not the bearing wear will be acceptably low.

The recommended decision process is outlined below.

Plot the values of accumulated wear versus time (H) for each bearing after Phase 2 and Phase 3 tests, namely.

Wear at H2 = 54 hour

Wear at H3 = 150 hours

Straight lines are then drawn through the plotted values of wear and extended to the right (See example Figure 1). If the extension intercepts the maximum acceptable value of wear (8 mils) at a value H less than the life goal for this pump, additional testing should be performed. If the intercept of the line with wear of 8 mils exceeds the life goal for this pump, no additional testing is required and bearing wear is acceptable. If additional testing is deemed necessary it should be done in a similar manner to that performed during Phase 3 with similar acceptance criteria and decision process. It is expected that such additional testing will either show a stable pump operation with no increase in bearing wear or increased bearing wear with unacceptable results.

2.6 Evaluation of Pump Acceptability

If bearing wear (after all testing phases) is acceptably low (as per decision process) and if vibration levels over the frequency spectrum of 3 cps to 5000 cps are acceptably low and show no unfavorable trend of increasing magnitude during the testing, the pump may be judged acceptable for its intended use.

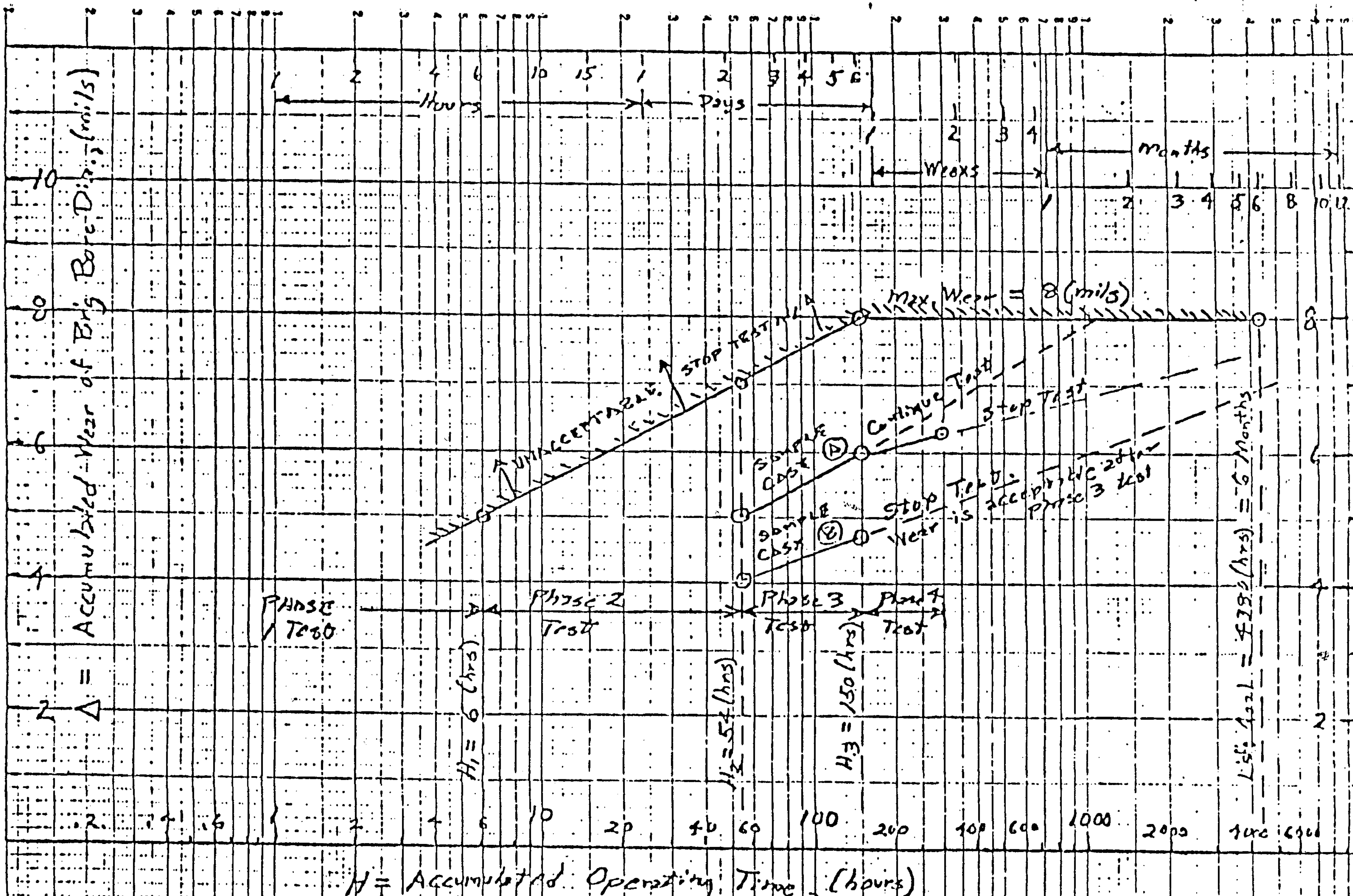

Figure 1 - Accumulated Wear of Brg Bore Diameter as a Function of Time for "Critical" Bearings

ENCLOSURE (6)

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 discussed in NUREG-0606. 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. Issues which have been resolved have been or are being incorporated into the NRC licensing guidance and are addressed as a part of the normal review process. However, we do request the information noted above for each of the issues listed below:

1. Waterhammer (A-1)
2. Steam Generator Tube Integrity (A-3)
3. ATWS (A-9)
4. Reactor Vessel Materials Toughness (A-11)
5. Steam Generator and Reactor Coolant Pump Support (A-12)
6. Systems Interaction (A-17)
7. Seismic Design Criteria (A-40)
8. Containment Emergency Sump Performance (A-43)
9. Station Blackout (A-44)
10. Shutdown Decay Heat Removal Requirements (A-45)
11. Seismic Qualification of Equipment in Operating Plants (A-46)
12. Safety Implications of Control Systems (A-47)
13. Hydrogen Control Measures and Effects of Hydrogen Burns on Safety Equipment (A-48)

Attached is a copy of the Generic Issues Branch SER contribution for a recent PWR plant, Virgil C. Summer. It is provided for your information only to assist you in your response regarding the Watts Bar Nuclear Plant, Units 1 and 2.

ATTACHMENT TO ENCLOSURE (6)

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

April 8, 1981

Docket No. 50-395

MEMORANDUM FOR: A. Schwencer, Chief
Licensing Branch #2, DL

FROM: Karl Kniel, Chief
Generic Issues Branch, DST

SUBJECT: SER INPUT: VIRGIL C. SUMMER, UNIT NO. 1

Plant Name: Virgil C. Summer, Unit No. 1
Docket Number: 50-395
Licensing Stage: OL
Responsible Branch and Project Manager, LB#2, W. F. Kane
DST Branch Involved: Generic Issues Branch
Description of Review: Unresolved Safety Issues
Requested Completion Date: March 24, 1981
Review Status: Complete

The Generic Issues Branch, DST, input to the Virgil C. Summer Unit No. 1 Safety Evaluation Report is enclosed. This appendix to the SER addresses the status of Unresolved Safety Issues pertaining to these facilities, and is in response to the ALAB-444 decision on this subject. That decision specified that "...each SER should contain a summary description of those generic problems under continuing study which have both relevance to facilities of the type under review and potentially significant public safety implications."

Page 5 of this Appendix references NUREG reports providing proposed generic resolution to five of the Unresolved Safety Issues. The Summer SER/SER Supplement section discussing the plant specific implementation of the generic programs is not available at this time and should be supplied by the Summer Project Manager when available. The Project Manager should also assure that plant specific implementation of resolved USIs is addressed in the body of the SER.

Karl Kniel

Karl Kniel, Chief
Generic Issues Branch
Division of Safety Technology

Enclosure:
Input to SER

cc: w/enclosure
See next page

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 issues are sometimes called "generic safety issues" because they are related to a particular class or type of nuclear facility rather than a specific plant. These issues have also been referred to as "unresolved safety issues." 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 the view of the Appeal Board, (pp. 25-29)

> "The responsibilities of a licensing board in the radiological health and safety sphere are not confined to the consideration and

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

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

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

"In short, the board (and the public as well) should be in a position to ascertain from the SER itself--without the need to resort to extrinsic documents--the staff's perception of the nature and extent of the relationship between each significant unresolved generic safety question and the eventual operation of the reactor under scrutiny. Once again, this assessment might well have a direct bearing upon the ability of the licensing board to make the safety findings required of it on the construction permit level even though the generic answer to the question remains in the offing. Among other things, the furnished information 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"

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

The Joint Explanatory Statement of the House-Senate Conference Committee for the 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 actions taken directly or indirectly to develop and implement corrective measures; (2) further actions planned concerning such measures; and (3) timetables and cost estimates of such actions. The Commission should indicate the priority it has assigned to each issue, and the basis on which priorities have been assigned."

In response to the reporting requirements of the new Section 210, the NRC staff submitted to Congress on January 1, 1978, a report, 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.

This 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 accepable 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.

"UNRESOLVED SAFETY ISSUES" (APPLICABLE TASK NOS.)

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." Eight of the 22 tasks identified with the "Unresolved Safety Issues" are not applicable to Virgil C. Summer Nuclear Station, Unit 1 and six of these eight tasks (A-6, A-7, A-8, A-39, A-10 and A-42) are peculiar to boiling water reactors. Tasks A-4 and A-5 address steam generator tube problems in Combustion Engineering and Babcock and Wilcox plants. With regard to the remaining 14 tasks that are applicable to this facility, the NRC staff has issued NUREG reports providing its proposed resolution of five of these issues. Each of these have been addressed in this Safety Evaluation Report or will be addressed in a future supplement. The table below lists those issues and the section of this Safety Evaluation Report in which they are discussed.

Task Number	NUREG Report and Title	Safety Evaluation Report Section
A-2	NUREG-0609, "Asymmetric Blowdown Loads on PWR Primary Systems"	3.9.3
A-24	NUREG-0588, "Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment"	7.7.2
A-26	NUREG-0224, "Reactor Vessel Pressure Transient Protection for Pressurized Water Reactors" and RSB BTP 5-2	5.4.2
A-31	Regulatory Guide 1.139, Guidance for Residual Heat Removal" and RSB BTP 5-1	Will be addressed in a future supplement.

Task Number	NUREG Report and Title	Safety Evaluation Report Section
A-36	NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants"	9.2.4

The remaining issues applicable to this facility are listed in the following table:

GENERIC TASKS ADDRESSING UNRESOLVED SAFETY ISSUES
THAT ARE APPLICABLE TO THE VIRGIL C. SUMMER NUCLEAR STATION, UNIT 1

1. A-1 Waterhammer
2. A-3 Westinghouse Steam Generator Tube Integrity
3. A-9 Anticipated Transients Without Scram
4. A-11 Reactor Vessel Materials Toughness
5. A-12 Potential for Low Fracture Toughness and Lamellar Tearing on PWR Steam Generator and Reactor Coolant Pump Supports
6. A-17 Systems Interactions in Nuclear Power Plants
7. A-40 Seismic Design Criteria
8. A-43 Containment Emergency Sump Reliability
9. A-44 Station Blackout

With the exception of Tasks A-9, A-43, and A-44, Task Action Plans for the generic tasks above are included in NUREG-0649, "Task Action Plans for Unresolved Safety Issues Related to Nuclear Power Plants." A technical resolution for Task A-9 has been proposed by the NRC staff in Volume 4 of NUREG-0460, issued for comment. This served as a basis for the staff's proposal for rulemaking on this issue. The Task Action Plan for Task A-43 was issued in January 1981, and the Task Action Plan for A-44 was issued in July 1980. Draft NUREG-0577 which represents staff resolution of USI A-12 was issued for comment in November 1979. The Draft NUREG contained the Task Action Plan for A-12. The information provided in NUREG-0649 meets most of the informational requirements of ALAB-444. Each Task Action Plan provides a description of the problem; the staff's approaches to its resolution; a general discussion of the bases upon which continued plant licensing or operation can proceed pending completion of the task; the technical organizations involved in the task and estimates of the manpower required; a description of the interactions with other NRC offices, the Advisory Committee on Reactor Safeguards and outside organizations; estimates of funding required for contractor supplied technical assistance; prospective dates for completing the task; and a description of potential problems that could alter the planned approach on schedule.

In addition to the Task Action Plans, the staff issues the "Office of Nuclear Reactor Regulation Unresolved Safety Issues Summary, Aqua Book" (NUREG-0606) on a quarterly basis which 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 nine "Unresolved Safety Issues" listed above as they relate to Virgil C. Summer Nuclear Station, Unit 1. Discussion of each of these issues including references to related discussions in the Safety Evaluation Report are provided below in Section C.5. Based on our review of these items, we have concluded, for the reasons set forth in Section C.5, that there is reasonable assurance that this facility 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 Reqcirements

A-46 Seismic Qualification of Equipment in Operating Plants

A-47 Safety Implications 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 present 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 Virgil C. Summer, Unit 1 are discussed in Section C.5.

C.5 Discussion of Tasks as they Relate to Virgil C. Summer Nuclear Station, Unit 1

A-1 Waterhammer

Waterhammer events are intense pressure pulses in fluid systems caused by any one of a number of mechanisms and system conditions.

Since 1971 there have been over 100 incidents involving waterhammer in pressurized water reactors and boiling water reactors. The waterhammers have involved steam generator feedrings and piping, decay heat removal systems, emergency core cooling systems, containment spray lines, service water lines, feedwater lines and steam lines. However, the systems most frequently affected by waterhammer effects are the feedwater systems. The most serious waterhammer events have occurred in the steam generator feedrings of pressurized water reactors. These types of waterhammer events are addressed in Section 10.4.3 of this Safety Evaluation Report.

With regard to protection against other 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, as described in Section 3.9.2 of this Safety Evaluation Report, we require that the applicant conduct a preoperational vibration dynamic effects test program in accordance with Section III of the ASME Code for all ASME Class 1 and Class 2 piping systems and piping restraints during startup and initial operation. These tests will provide adequate assurance that the piping and piping restraints have been designed to withstand dynamic effects due to valve closures, pump trips and other operating modes associated with the design operational transients.

Nonetheless, in the unlikely event that a large pipe break did result from a severe waterhammer event, core cooling is assured by the emergency core cooling systems described in Section 6.3 of this Safety Evaluation Report and protection against the dynamic effects of such pipe breaks inside and outside of containment is provided as described in Section 3.6 of this Safety Evaluation Report.

Task A-1 may identify some potentially significant waterhammer scenarios that have not explicitly been accounted for in the design and operation of nuclear power plants. The task has not as yet identified the need for requiring any additional measures beyond those already required in the short term.

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

A-3 Westinghouse Steam Generator Tube Integrity

The primary concern is the capability of steam generator tubes to maintain their integrity during normal operation and postulated accident conditions. In addition, the requirements for increased steam generator tube inspections and repairs have resulted in significant increases in occupational exposures to workers. Corrosion resulting in steam generator tube wall thinning (wastage) has been observed in several Westinghouse plants for a number of years. Plants operating exclusively with an all volatile secondary water treatment process have not experienced this form of degradation to date. Another major corrosion-related phenomenon has also been observed in a number of plants in recent years, resulting from a buildup of support plate corrosion products in the annulus between the tubes and the support plates. This buildup eventually causes a diametral reduction of the tubes, called "denting," and deformation of the tube support plates. This phenomenon has led to other problems, including stress corrosion cracking, leaks at the tube/support plate intersections, and U-bend section cracking of tubes which were highly stressed because of support plate deformation.

Specific measures such as steam generator design features and a secondary water chemistry control and monitoring program, that the applicant has employed to minimize the onset of steam generator tube problems are described in Section _____ of this Safety Evaluation Report. In addition, Section _____ of this Safety Evaluation Report discusses the inservice inspection requirements. As described in Section _____, the applicant has met all current requirements regarding steam generator tube integrity. The Technical Specification will include requirements for actions to be taken in the event that steam generator tube leakage occurs during plant operation.

Task A-3 is expected to result in improvements in our current requirements for inservice inspection of steam generator tubes. These improvements will include a better statistical basis for inservice inspection program requirements and consideration of the cost/benefit of increased inspection. Pending completion of Task A-3, the measures taken at this facility should minimize the steam generator tube problems encountered. Further the inservice inspection and Technical Specification requirements will assure that the applicant and the NRC staff are alerted to tube degradation should it occur. Appropriate actions such as tube plugging, increased and more frequent inspections and power derating could be taken if necessary. Since the improvements that will result from Task A-3 will be procedural, i.e., an improved inservice inspection program, they can be implemented by the applicant after operation of this facility begins, if necessary.

Based on the foregoing, we have concluded that this facility 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.

The anticipated transient without scram issue and the requirements that must be met by the applicant prior to operation of the facility are discussed in Section 15.3.5 of this Safety Evaluation Report.

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

Based on our review, we have concluded that there is reasonable assurance that this facility 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, a rapidly propagating catastrophic failure mode for a component containing flaws, is described quantitatively by a material property generally denoted as "fracture toughness." Fracture toughness has different values and charactersitics 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 materials. The effect of neutron radiation on the fracture toughness of the vessel material is accounted for in developing and revising these Technical Specification limitations.

For the service times and operating conditions typical of current operating plants, reactor vessel fracture toughness for most plants provides adequate margins of safety against vessel failure under operating, testing, maintenance, and anticipated transient conditions, and accident conditions over the life of the plant. However, results from a reactor vessel surveillance program and analyses performed for up to 20 older operating pressurized water reactors and those for some more recent vintage plants will have marginal toughness, relative to required margins at normal full power after comparatively short periods of operation. In addition, results from analyses performed by pressurized water reactor manufacturers indicate that the integrity of some reactor vessels may not be maintained in the event that a main steam line break of a loss-of-coolant accident occurs after approximately 20 years of operation. The principal objective of Task A-11 is to develop an improved engineering method and safety criteria to allow a more precise assessment of the safety margins that are available during normal operation and transients in older reactor vessels with marginal fracture toughness and of the safety margins available during accident conditions for all plants.

Based on our evaluation of this facility's reactor vessel materials toughness, we have concluded that this unit will have adequate safety margins against brittle failure during operating, testing, maintenance and anticipated transient conditions over the life of the units. Since Task A-11 is projected to be completed well in advance of this facility's reactor vessel reaching a fluence level which would noticably reduce fracture resistance, acceptable vessel integrity for the postulated accident conditions will be assured at least until the reactor vessel is reevaluated for long-term acceptability.

In addition, the surveillance program required by 10 CFR 50, Appendix H will afford an opportunity to reevaluate the fracture toughness periodically during the first half of design life.

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

A-12 Fracture Toughness of Steam Generator and Reactor Coolant Pump Supports

During the course of the licensing action for North Anna Power Station Unit No. 1 and 2 a number of questions were raised as to the potential for lamellar tearing and low fracture toughness of the steam generator and reactor coolant pump support materials for those facilities. Two different steel specifications (ASTM A36-70a and ASTM A572-70a) covered most of the material used for these supports. Toughness tests, not originally specified and not in the relevant ASTM specifications, were made on those heats for which excess material was available. The toughness of the A36 steel was found to be adequate, but the toughness of the A572 steel was relatively poor at an operating temperature of 80°F.

Since similar materials and designs have been used on other nuclear plants, the concerns regarding the supports for the North Anna facilities are applicable to other PWR plants. It was therefore necessary to reassess the fracture toughness of the steam generator and reactor coolant pump support materials for all operating PWR plants and those in CP and OL review.

NUREG-0577, "Potential for Low Fracture Toughness and Lamellar Tearing on PWR Steam Generator and Reactor Coolant Pump Supports," was issued for comment in November 1979. This report summarizes work performed by the NRC staff and its contractor, Sandia Laboratories, in the resolution of this generic activity. The report describes the technical issues, the technical studies performed by Sandia Laboratories, the NRC staff's technical positions based on these studies, and the NRC staff's plan for implementing its technical positions. As a part of initiating the implementation of the findings in this report, letters were sent to all applicants and licensees on May 19 and 20, 1980. In these letters a revised proposed implementation plan was presented and specific criteria for material qualifications were defined.

Many comments on both the draft of NUREG-0577 and the letters of May 19 and 20 have been received by the NRC staff and detailed consideration is presently being given to these comments. After completing our review and analysis of the comments provided, we will issue the final revision of NUREG-0577 which will include a full discussion and resolution of the comments and a final plan for implementation.

We estimate that our implementation review will require approximately two years. Since many factors (initiating event, low fracture toughness in a critical support member in tension, low operating temperature, large flaw) must be simultaneously present for failure of the support system we have determined that licensing for pressurized water reactors should continue during the implementation phase. Our conclusions regarding licensing and subsequent operation are not sensitive to the estimated length of time required for this work.

A-17 Systems Interaction in Nuclear Power Plants

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

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

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

The NRC staff's current review procedures assign primary responsibility for review of various technical areas and safety systems to specific 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 provided 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 of negating or seriously degrading the performance of safety functions. The study concentrated on common cause on 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 was not within the Phase I scope of the Sandia Study.

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

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

A-40 Seismic Design Criteria - Short-Term Program

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

As discussed in Section 3.7 of this Safety Evaluation Report the seismic design basis and seismic design of the facility have been evaluated at the operating license stage and have been found acceptable. We do not expect the results of Task A-40 to affect these conclusions because the techniques under consideration are essentially those utilized in the review of this facility. Should the resolution of Task A-40 indicate a change is needed in licensing requirements, all operating reactors, including Summer will be reevaluated on a case-by-case basis. Accordingly, we have concluded that this facility can be operated prior to the 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 emergency sump at the low point in the containment. This water would be recirculated through the reactor system by the emergency core cooling pumps to maintain core cooling. This water would also be circulated through the containment spray system to remove heat and fission products from the containment. Loss of the ability to draw water from the emergency sump could disable the emergency core cooling and containment spray systems.

One postulated means of losing the ability to draw water from the emergency sump could be blockage by debris. A principal source of such debris could be the thermal insulation on the reactor coolant system piping. In the event of a piping break, the subsequent violent release to the high pressure water in the reactor coolant system could rip off the insulation in the area of the break. This debris could then be swept into the sump, potentially causing blockage.

Currently, regulatory positions regarding sump design are presented in Regulatory Guide 1.82, "Sumps for Emergency Core Cooling and Containment Spray Systems," which address debris (insulation). Regulatory Guide 1.82 recommends, in addition to providing redundant separated sumps, that two protective screens be provided. A low approach velocity in the vicinity of the sump is required to allow insulation to settle out before reaching the sump screening; and it is required that the sump remain functional assuming that one-half of the screen surface area is blocked.

A second postulated means of losing the ability to draw water from the emergency sump could be abnormal conditions in the sump or at the pump inlet such as air entrainment, vortices, or excessive pressure drops. These conditions could result in pump cavitation, reduced flow and possible damage to the pumps.

Currently, regulatory positions regarding sump testing are containedin Regulatory Guide 1.79, "Preoperational Testing of Emergency Core Cooling Systems for Pressurized Water Reactors," which addresses the testing of the recirculation function. Both in-plant and scale model tests have been performed by applicants to demonstrate that circulation through the sump can be reliably accomplished.

As indicated in Section 6.3.3 of this Safety Evaluation Report, the applicant will perform out-of-plant scale model tests of the containment sump design. The applicant will be required to demonstrate that there is reasonable assurance that the sump design will perform as expected following a loss-of-coolant accident.

The near term implementation of Task A-43 for this facility is expected to be procedural in nature and assure adequate housekeeping and emergency procedures to supplement the sump tests discussed above. Accordingly, we have concluded that this facility can be operated prior to ultimate resolution of this generic issue without endangering the health and safety of the public.

A-44 Station Blackout

Electrical power for safety systems at nuclear power plants must be supplied by at least two redundant and independent divisions. The systems used to remove decay heat to cool the reactor core following a reactor shutdown are included among the safety systems that must meet these requirements. Each electrical division for safety systems includes an offsite alternating current power connection, 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., 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 unavilable. 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 Summer facility. Nonetheless, a combination of design, operation and testing requirements that have been imposed on the applicant will assure that these units will have substantial resistance to a loss of all alternating current and that, even if a loss of all alternating current should occur, there is reasonable assurance that the core will be cooled. These 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 this Safety Evaluation Report.

If offsite power is lost, two diesel generators and their associated distribution systems will deliver emergency power to safety-related equipment. Our review of the design, testing, surveillance, and maintenance provisions for the onsite emergency diesels is described in Section 8.3 of the SER. Our requirements include preoperational testing to assure the reliability of the installed diesel generators in accordance with our requirements discussed in the SER. In addition, the applicant has been requested to implement a program for enhancement of diesel generator reliability to better assure the long-term reliability of the diesel generators. This program resulted from recommendations of NUREG/CR-0660, "Enhancement of Onsite Emergency Generator Reliability."

Event if both offsite and onsite alternating current power are lost, cooling water can still be provided to the steam generators by the auxiliary feedwater system by employing a steam turbine driven pump that does not rely on alternating current power for operation. Our review of the auxiliary feedwater system design and operation is described in Section _____ of the Safety Evaluation Report.

The issue of station blackout was also considered by the Atomic Safety and Licensing Appeal Board (ALAB-603) for the St. Lucie Unit No. 2 facility. In addition, in view of the completion schedule for Task A-44 (October 1982), the Appeal Board recommended that the Commission take expeditious action to ensure that other plants and their operators are equipped to accommodate a station blackout event. The Commission has reviewed this recommendation and determined that some interim measures should be taken at all facilities including Summer 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 required. The staff notified the applicant of these requirements in a letter from D. Eisenhut, NRC, to the applicant dated February 25, 1981. We will condition the operating license for Summer that their procedures and training be completed by fuel load date.

Based on the above, we have concluded that there is reasonable assurance that Summer 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

Under normal operating conditions, power generated within a reactor is removed as steam to produce electricity via a turbine generator. Following a reactor shutdown, a reactor produces insufficient power to operate the turbine; however, the radioactive decay of fission products continues to produce heat (so-called "decay heat"). Therefore, when reactor shutdown occurs, other measures must be available to remove decay heat from the reactor to ensure that high temperatures and pressures do not develop which could jeopardize the reactor and the reactor coolant system. It is evident, therefore, that all light water reactors (LWRs) share two common decay heat removal functional requirements: (1) to provide a means of transferring decay heat from the reactor coolant system to an ultimate heat sink and (2) maintain sufficient water inventory inside the reactor vessel to ensure adequate cooling of the reactor fuel. The reliability of a particular power plant to perform these functions depends on the frequency of initiating events that require or jeopardize decay heat removal operations and the probability that required systems will respond to remove the decay heat.

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 significant reduce the overall risk to the public.

The primary method for removal of decay heat from pressurized water reactors is via the steam generators to the secondary system. This energy is transferred on the secondary side to either the main feedwater or auxiliary feedwater systems, and it is rejected to either the turbine condenser or the atmosphere via the steamline safety/relief valves. Following the TMI-2 accident, the importance of the auxiliary feedwater system was highlighted and a number of steps were taken to improve the reliability of the auxiliary feedwater system. The staff's review of these items is contained in Section ____ of this Safety Evaluation Report. It was also stipulated that plants must be capable of providing the required AFW flow for at least two hours from one auxiliary feedwater pump train, independent of any alternating current power source (that is, if both off-site and on-site alternating current power sources are lost).

Pressurized water reactors also have alternate means of removing decay heat if an extended loss of feedwater is postulated. This method is known as "feed and bleed" and uses the high pressure injection system to add water coolant (feed) at high pressure to the primary system. The decay heat increases the system pressure and energy is removed through the power-operated relief valves and/or the safety valves (bleed), if necessary.

At low primary system pressure (below about 200 psi), the long-term decay heat is removed by the residual heat removal system to achieve cold shutdown conditions.

Based on the foregoing, we have concluded that Virgil C. Summer, Unit 1 can be operated prior to ultimate resolution of this generic issue without endangering the health and safety of the public.

A-46 Seismic Qualification of Equipment in Operating Plants

The design criteria and methods for the seismic qualification of mechanical and electrical equipment in nuclear power plants have undergone significant change during the course of the commercial nuclear power program.

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

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

A-47 Safety Implications of Control Systems

This issue concerns the potential for transients or accidents being made more severe as a result of control system failures or malfunctions. These failures or malfunctions may occur independently or as a result of the accident or transient under consideration. One concern is the potential for a single failure such as a loss of a power supply, short circuit, open circuit, or sensor failure to cause simultaneous malfunction of several control features. Such an occurrence 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 could 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. It is generally believed by the staff that such control system failures would not lead to serious events or result in conditions that safety systems cannot safely handle. Systematic evaluations have not been rigourously performed to verify this belief. The potential for an accident that could 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 evaluations are required. The purpose of this Unresolved Safety Issue is to define generic criteria that will be used for plant-specific evaluations.

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

A systematic evaluation of the control system design, 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 such as steam generator overfill and overcooling events would be adequately mitigated by the safety systems. In addition, systematic reviews of safety systems have been performed with the goal of ensuring that control system failures (single or multiple) will not defeat safety system action.

Based on the above, we have concluded that there is reasonable assurance that the Summer Unit 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 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 the General Design Criteria 41, "Containment Atmosphere Cleanup" in Appendix A to 10 CFR Part 50 require that systems be provided to control hydrogen concentrations in the containment atmosphere following a postulated accident to ensure that containment integrity is maintained.

10 CFR Section 50.44 requires that the combustible gas control system provided be capable of handling the hydrogen generated as a result of degradation of the emergency core cooling system such that the hydrogen release is five times the amount calculated in demonstrating compliance with 10 CFR Section 50.46 or the amount corresponding to reaction of the cladding to a depth of 0.00023 inch, whichever amount is greater.

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

Recognizing that a number of years may be required to complete this rulemaking proceeding, a set of short-term or interim actions relative to hydrogen control requirements were developed and implemented. These interim measures were described in a second October 2, 1980 Federal Register notice. For plants with large dry containments such as Virgil C. Summer, Unit 1, no near-term mitigation measures are required by the interim rule.

The Virgil C. Summer plant has about two million cubic feet of net free volume. Assuming 30 to 50% metal-water reaction in the core, the resulting uniformly mixed concentration of hydrogen in the containment will range from 6 to 10%. This is well below the concentrations for detonation and even below the limits for combustion if there were more than 50% steam in the containment atmosphere.

Design pressure of the Virgil C. Summer plant is 57 psig. Analyses performed on the Zion and Indian Point plants show that the failure pressures are greater than twice the design pressures.

If the substantial amount of metal-water reaction were to occur shortly following onset of a large LOCA and while the containment is still near its peak pressure, the pressure increase caused by the noncondensible hydrogen gas and its associated exothermic formation energy will be substantially less than the failure pressure. If the metal-water reaction were to occur well after onset of the large LOCA, then the containment heat removal system would have condensed much of the steam in the containment and reduced the containment pressure. This would provide a substantial margin for accommodating the hydrogen generated by the metal-water reaction.

In addition, the "Short Term Lessons Learned" from the TMI-2 accident have been implemented on the Virgil C. Summer plant. This action will reduce the likelihood of accidents that could lead to substantial amounts of metal-water reaction.

Accordingly, pending resolution of this Unresolved Safety Issue and the rulemaking proceeding on hydrogen generation, the Virgil C. Summer plant can be operated without undue risk to the health and safety of the public.

ENCLOSURE (7)

REQUEST FOR ADDITIONAL INFORMATION

362.44 The Giles County Virginia earthquake of 1897 is the controlling earthquake for the seismic design of nuclear plants in the Southern Valley and Ridge tectonic province. Watts Bar Nuclear Plant is located in this province.

Dr. G. A. Bollinger has been conducting research on the Giles County, Virginia seismic zone. He has recently written a report titled "The Giles County, VA Seismic Zone - Configuration and Hazard Assessment" which is to be presented at a conference in September, 1981.

Based on the local seismic activity Dr. Bollinger implies the existence of a buried fault in the Giles County area. He uses the largest extent of the seismic zone, taking into account errors in hypocenter location, in order to calculate a possible maximum earthquake of surface wave magnitude $M_S = 7$ for this zone.

Provide a discussion on any effect this hypothesis has on the following with respect to the Watts Bar Plant:

a) The potential of the 1897 earthquake being associated with this specific geologic structure;

b) The potential of an earthquake up to $M_S = 7.0$ located in Giles County, and any far field ground motion effect (both peak values and response spectrum) at the site from an $M_S = 7.0$ event located in Giles County;

c) The potential of similar seismogenic structures being located near the Watts Bar site, and any effects at the site from earthquakes on these seismogenic structures.