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Docket Nos.: STN 50-454
and STN 50-455

Mr. Louis O. DelGeorge
Director of Nuclear Licensing
Commonwealth Edison Company
Post Office Box 767
Chicago, Illinois 60690



Dear Mr. DelGeorge:

Subject: Review of Unresolved Safety Issues in the Byron SER

We request that you submit information regarding the status of Unresolved Safety Issues (USI) relating to Byron to supplement our generic discussion of these issues in the SER. This information should include a summary description of the relevant programs and the interim measures you have taken pending resolution of the issues. Enclosure 1 includes a list of USIs that are applicable to the Byron facility. We have also included Enclosure 2, the Generic Issues Branch SER contribution for a recent PWR plant to assist you in your response.

Prior to your formal reply to this request, we would like to meet with you briefly to discuss the purpose of this request and the response desired. Please inform us within sever days of receipt of this letter of your schedule for responding to this request and your availability for such a meeting. If you need further information, contact the Byron project manager.

Sincerely,

Original signed by:
B. J. Youngblood

B. J. Youngblood, Chief
Licensing Branch No. 1
Division of Licensing

Enclosures:
As stated

cc w/encls.: See next page



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| SURNAME | KLKiper/lg | WKane | BJYoungblood | | | | |
| DATE | 11/6/81 | 11/9/81 | 11/10/81 | | | | |

Mr. Louis O. DelGeorge
Director of Nuclear Licensing
Commonwealth Edison Company
Post Office Box 767
Chicago, Illinois 60690

ccs:

Mr. William Kortier
Atomic Power Distribution
Westinghouse Electric Corporation
P. O. Box 355
Pittsburgh, Pennsylvania 15230

Paul M. Murphy, Esq.
Isham, Lincoln & Beale
One First National Plaza
42nd Floor
Chicago, Illinois 60603

Mrs. Phillip B. Johnson
1907 Stratford Lane
Rockford, Illinois 61107

Ms. Bridget Little Rorem
Appleseed Coordinator
117 North Linden Street
Essex, Illinois 60935

Dr. Bruce von Zellin
Department of Biological Sciences
Northern Illinois University
DeKalb, Illinois 61107

Mr. Edward R. Crass
Nuclear Safeguards and Licensing Division
Sargent & Lundy Engineers
55 East Monroe Street
Chicago, Illinois 60603

Nuclear Regulatory Commission
Region III
Office of Inspection and Enforcement
799 Roosevelt Road
Glen Ellyn, Illinois 60137

Myron Cherry, Esq.
Cherry, Flynn and Kanter
1 IBM Plaza, Suite 4501
Chicago, Illinois 60611

U. S. Nuclear Regulatory Commission
Resident Inspectors Office
4448 German Church Road
Byron, Illinois 61010

Ms. Diane Chavez
602 Oak Street
Rockford, Illinois 61104

Enclosure 1

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)

# 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 acceptable over the lifetime of the plants it affects."

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

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

"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 Requirements

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 pumo 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 I 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.