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SCHWENCER,A. Licensing Branch 2

SUBJECT: Forwards revised Section 1.5 to FSAR re unresolved safety issues.Pages will be incorporated into Amend 23.

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January 11, 1982
G02-82-23
SS-L-02-KSN-82-001

Docket No. 50-397

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

Dear Mr. Schwencer:

Subject: NUCLEAR PROJECT NO. 2
UNRESOLVED SAFETY ISSUES

Reference: Letter, A. Schwencer to R.L. Ferguson, "WNP-2 FSAR - Request for Additional Information", dated November 16, 1981

Enclosed are sixty (60) copies of the revised Section 1.5, Unresolved Safety Issues, per the referenced letter requesting additional information. These pages will be incorporated into Amendment 23 of the WNP-2 FSAR.

Very truly yours,

GD Bouchey

G. D. Bouchey
Deputy Director, Safety and Security

KSN/jca
Enclosures

cc: R Auluck - NRC
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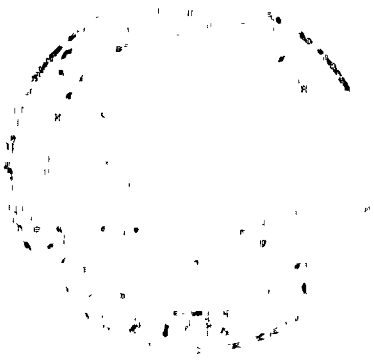

1.5 REQUIREMENTS FOR FURTHER TECHNICAL INFORMATION

1.5.1 DEVELOPMENT OF BWR TECHNOLOGY

1.5.1.1 ACRS Concerns

This section addresses those concerns of the Advisory Committee on Reactor Safeguards pertaining to WNP-2. Table 1.5-1 summarizes resolution of ACRS concerns.

1.5.1.1.1 Containment Design Features to Minimize Effects of Bypass Leakage

The provisions made in the design to minimize bypass leakage (directly from inside the containment to outside the reactor building) after a postulated loss-of-coolant accident are described in Section 6.2.3 of this FSAR.

1.5.1.1.2 Pipe Whip Protection Provisions

"The Committee believes that protection against pipe whip should be provided by the applicant in accordance with criteria being developed by the AEC Regulatory Staff."

Response:

The provisions made in the design of WNP-2 to provide protection against dynamic effects associated with the postulated rupture of piping (pipe whip) are described in Section 3.6 of this FSAR.

1.5.1.1.3 Design Criteria for Inactive Pumps and Valves

"Active pumps and valves of the reactor coolant pressure boundary required to perform safety functions will be designed to deformation limits for which the calculated primary stresses will be in the elastic range. Acceptable design criteria for inactive pumps and valves are yet to be established. This matter should be resolved in a manner satisfactory to the Regulatory Staff."

Response:

This item has been resolved and the resolution documented in the letter of April 23, 1974 from R.C. DeYoung to J.J. Stein.

1.5.1.1.4 Main Steam Line Leakage Control System

"The applicant has proposed to install a sealing system to ensure minimal leakage through the main steam line isolation valves following a postulated loss-of-coolant accident and has in progress a study to establish the design of such a system. The Committee believes that a sealing system should be installed. This matter should be resolved in a manner satisfactory to the Regulatory Staff prior to completion of construction of the plant."

Response:

The provisions made in the design of WNP-2 to minimize leakage through the main steam line isolation valves following a postulated loss-of-coolant accident are described in Section 6.7 of this FSAR.

1.5.1.1.5 Mitigation of Consequences of Control-Rod Drop Accident

"Analyses of postulated control-rod drop accidents occurring in similar cores during certain portions of the fuel cycle indicate unacceptable results. Studies of provisions to reduce the probability of this accident to negligible levels are underway. This matter should be resolved in a manner satisfactory to the Regulatory Staff prior to completion of construction."

Response:

The design and procedural provisions that are being used on this plant are described in Section 4.3.2.6. These provisions are adequate to control individual rod worths, and ensure the consequences of a postulated rod drop accident are acceptable.

1.5.1.1.6 Anticipated Transients Without Scram

"The applicant has studied design features to make tolerable the consequences of failure to scram during anticipated transients, and has concluded that automatic tripping of the recirculation pumps and injection of boron could provide for a suitable backup to the control-rod system for this type of event. The Committee believes that this approach represents a substantial improvement and should be provided for the Hanford No. 2 reactor. However, further evaluation of the sufficiency of this approach and the specific means of implementing the proposed pump trip should be made. This matter should be resolved in a manner satisfactory to the Regulatory Staff and the ACRS during construction of the plant."

Response:

The consequences of an anticipated transient without scram (ATWS) are mitigated by tripping the recirculation pumps and by manual insertion of the control rods. (For more information, see 15.8).

1.5.1.2 Current Development Program

1.5.1.2.1 Loose Parts Detection

A Loose Parts Detection System will be provided. See 7.7.1.12 for a system description.

~~1.5.1.2.2 Mark II Containment Suppression Pool Dynamic Loading~~

~~The Washington Public Power Supply System, in conjunction with other Mark II owner utilities, has submitted a design basis document designated as "Mark II Containment Dynamic Forcing Function Information Report" (DFFIR), NEDO-21061, and NEDE-21061P describing the suppression pool dynamic loading phenomena during a safety/relief valve actuation or LOCA event. The evaluation of that design basis document against the current WPPSS Nuclear Project No. 2 design was prepared and submitted to the NRC. A verification program to demonstrate the conservatism of the DFFIR has been sponsored by the Mark II owners and is described in the "Mark II Containment Supporting Program Report", GE Document NEDO-21297.~~

~~Additional information concerning plant evaluation for suppression pool dynamic loading is contained in the Plant Design Assessment Report for SRV and LOCA loads, Revision 2, transmitted to the NRC as Appendix G to the FSAR, September 19, 1979.~~

TABLE 1.5-1
RESOLUTION OF ACRS CONCERNS

ACRS Concern		Resolved*	Unresolved	Addressed in FSAR Section (Where Applicable)
ACRS Group I:				
1.	NPSH for ECCS Pumps	X		App. C, R.G. 1.1
2.	Emergency Power	X		8.3, 7.3.2.1.3
3.	Hydrogen Control	X		6.2.5
4.	Instrument Lines Penetrating Containment	X		7.1.2.4(a)
5.	Strong Motion Seismic Instrumentation	X		3.7.4
6.	Fuel Pool Design Basis	X		9.1.2.3, 9.1.3.3
7.	Pump Flywheel Missiles	X		Not Applicable
8.	Protection Against Industrial Sabotage	X		13.6
9.	Vibration Monitoring	X		1.5.1.2.1, 3.9.2, App. C, R.G. 1.20
10.	Inservice Inspection of RCPB	X		5.2.4
11.	Quality Assurance During Design, Construction and Operation	X		17
12.	Inspection of BWR Steam Lines Beyond Isolation Valves	X		6.6
13.	Independent Check of Primary System Stress Analysis	X		Primary System design to ASME BPV Code Section 3. Section 5.2

TABLE 1.5-1 (Continued) Page 2 of 7

ACRS Concern		Resolved*	Unresolved	Addressed in FSAR Section (Where Applicable)
Group I (continued):				
14.	Operational Stability of Jet Pumps	X		Confirmed at Dresden 2 and 3
15.	Pressure Vessel Surveillance of Fluence and NDT Shift	X		5.3.1
16.	Nil Ductility Properties of Pressure Vessel Materials	X		5.3.1
17.	Operation of Reactor with less than all loops in service	X		Addressed in Millstone 1, Docket No. 50-245
18.	Criteria for Pre-operational Testing	X		14.1
19.	Diesel Fuel Capacity	X		9.5.4
20.	Biological Shield Capability	X		3.8.3.4
21.	Operating One Plant while others are under Construction	X		Not Applicable
22.	Seismic Design of Steam Lines	X		3.2.1, 3.7.2.1.8.2
23.	Quality Group Classification of Pressure Retaining Components	X		3.2.2, 3.2.3
24.	Ultimate Heat Sink	X		1.2.2.12.3, 9.2.5

TABLE 1.5-1 (Continued) Page 3 of 7

ACRS Concern		Resolved*	Unresolved	Addressed in FSAR Section (Where Applicable)
Group I (continued):				
25.	Instrumentation to Detect Stresses in Containment Walls	X		3.8.2.7, 7.5.1.5
Group IA:				
1.	Use of Furnace sensitized stainless steel	X		5.2.3
2.	Primary system detection and location of leaks	X		7.6.1.3
3.	Protection against pipe whip	X		3.6
4.	Anticipated transients without scram	X		Will be addressed in response to WASH-1270
5.	ECCS capabilities of current and older plants	X		6.3. Conformance to 10 CFR 50, Appendix K addressed in 6.3.3
Group IB:				
1.	Positive moderator coefficient	X		4.3
2.	Fixed in-core detectors on high power PWRs	X		Not Applicable
3.	Performance of critical components in post-LOCA environment	X		7.5.2, NEDO-10698

TABLE 1.5-1 (Continued) Page 4 of 7

ACRS Concern		Resolved*	Unresolved	Addressed in FSAR Section (Where Applicable)
Group IB (continued):				
4.	Vacuum relief valves controlling bypass paths on BWR pressure suppression containments	X		6.2.1
5.	Emergency power for two or more reactors at same site	X		8
6.	Effluents from light water cooled reactors	X		11.3
7.	Control rod ejection accident	X		Not Applicable
Group IC:				
1.	Main steam isolation valve leakage of BWRs	X		App. C, R.G. 1.96, 5.4.5
2.	Fuel densification	X		Topical report NEDM-10735
3.	Rod sequence control system	X		7.7.1.11 NEDO-10527
4.	Seismic Category I requirements for auxiliary systems	X		9.1.3, 11.3.1.3, 11.5
Group II:				
1.	Turbine missiles		X	3.5.1.3
2.	Containment Sprays	X		6.5.2

TABLE 1.5-1 (Continued) Page 5 of 7

ACRS Concern	Resolved*	Unresolved	Addressed in FSAR Section (Where Applicable)
Group II (continued):			
3. Pressure Vessel Thermal Shock, Post-LOCA		X	5.3.1, 5.3.3.6
4. Instruments to Detect Fuel Failure	X		7.6.1.1
5. Loose Parts Monitoring		X	3.9.2, 4.4.6.1
6. Common Mode Failures	X		7.1.2, Appendix Topical Report NEDO-10189
7. Fuel Behavior Under Abnormal Conditions		X	4.2, 4.3, 4.4, 15.0, Topical Reports NEDO-10174, NEDO-10179 NEDO-10208, NEDO-10505 and NEDM-10735
8. BWR Recirculation Pump Overspeed During LOCA	X		5.4.1.4
9. Seismic Scram	X		No seismic scram is incorporated. Seismic instrumentation meets R.G. 1.12 (App. C).
10. ECCS capability for Future Plants		X	6.3, 15.0 Topical Reports NEDO-10892, NEDO-10179.

TABLE 1.5-1 (Continued) Page 6 of 7
RESOLUTION OF ACRS CONCERNS

ACRS Concern	Resolved*	Unresolved	Addressed in FSAR Section (Where Applicable)
Group II (continued):			
11. Instrumentation to Follow Course of Accident		X	7.5.1, 7.5.2
Group IIA:			
1. Pressure in Containment Following LOCA	X		6.2.1.3, Topical Report NEDO-10320, NEDO-20345, NEDO-20550, NEDO-20533, NEDM-10976, NEDO-10329
2. BWR Control Rod Drop Accident	X		15.4.9
3. Ice Condenser Containments			Not Applicable
4. Rupture of High Pressure Lines Outside Containment	X		3.6.1, 15.0
5. PWR Pump Overspeed			Not Applicable
6. Isolation of Low Pressure from High Pressure Systems	X		5.2, 6.2.4.3, 6.3.2.2, 7.3
7. Steam Generator Tube Leaks			Not Applicable
8. ACRS/NRC Periodic 10 Year Review of Older Reactors			Not Applicable

TABLE 1.5-1 (Continued) Page 7 of [illegible]
RESOLUTION OF ACRS CONCERNS

ACRS Concern		Resolved*	Unresolved	Addressed in FSAR Section (Where Applicable)
Group IIB:				
1.	Hybrid Reactor Protection Systems		X	7.2.1.1, 7.2.2
2.	8x8 BWR Fuel Qualification	X		4.2
3.	Behavior of Mark III Containment		X	Not Applicable
4.	Stress Corrosion Cracking		X	3.1.2.4.3, 5.2.3.4, 5.2.4
Group IIC:				
1.	Locking Out ECCS Power Operated Valves		X	7.3.1.1.1
2.	Fire Protection		X	8.3.3, 9.5.1
3.	Design Features to Control Sabotage		X	13.6
4.	Decontamination and Decommissioning of Reactors		X	Not Applicable
5.	Vessel Supporting Structures		X	3.9.1, 3.9.3, 3.9.5
6.	Water Hammer			6.3.2.2.5
7.	Maintenance and Inspection of Plants		X	5.2.4, 3.9.6, 3.8.2.7
8.	Behavior of Mark I Containments	X		Not Applicable

* - Resolved means a specified conclusion or policy decision has been reached by the NRC and ACRS.

1.5 REQUIREMENTS FOR FURTHER TECHNICAL INFORMATION

1.5.1 UNRESOLVED SAFETY ISSUES

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

In some cases, immediate action is taken to assure safety, e.g., the derating of boiling water reactors as a result of the channel box wear problems in 1975. In other cases, interim measures, such as modifications to operating procedures, may be sufficient to allow further study of the issue prior to making licensing decisions. In most cases, however, the initial assessment indicates that immediate licensing actions or changes in licensing criteria are not necessary. In any event, further study may be deemed appropriate to make judgements as to whether existing NRC staff requirements should be modified to address the issue for new plants or if backfitting is appropriate for the long-term operation of plants already under construction or in operation.

These issues are sometimes called "generic safety issues" because they are related to a particular class or type of nuclear facility rather than a specific plant. Certain of these issues have been designated as "unresolved safety issues". (NUREG-0410, "NRC Program for the Resolution of Generic Issues Related to Nuclear Power Plants", dated January 1, 1978.) However, as discussed above, such issues are considered on a generic basis only after the NRC 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.

1.5.1.1 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.

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

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, "Unresolved Safety Issues Plan".

In response to the reporting requirements of the new Section 210, the NRC staff submitted to Congress on January 1, 1978, a report describing the NRC generic issues program (NUREG-0410). 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. In 1978, the NRC undertook a review of over 130 generic issues addressed in the NRC program to determine which cases 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 the final approval by the NRC Commissioners.

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

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

Further, the report indicates that in applying this definition, matters that pose "important questions concerning the adequacy of existing safety requirements" were judged to be those for which resolution is necessary to (1) compensate for a possible major reduction in the degree of protection of the public health and safety, or (2) provide a potentially significant

decrease in the risk to the public health and safety. Quite simply, an "Unresolved Safety Issue" is potentially significant from a public safety standpoint and its resolution is likely to result in NRC action on the affected plants.

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

"UNRESOLVED SAFETY ISSUES" (APPLICABLE TASK NOS.)

1. Water Hammer - (A-1)
2. Asymmetric Blowdown Loads on the Reactor Coolant System - (A-2)
3. Pressurized Water Reactor Steam Generator Tube Integrity - (A-3, A-4, A-5)
4. BWR Mark I and Mark II Pressure Suppression Containments - (A-6, A-7, A-8, A-39)
5. Anticipated Transients Without Scram - (A-9)
6. BWR Nozzle Cracking - (A-10)
7. Reactor Vessel Materials Toughness - (A-11)
8. Fracture Toughness of Steam Generator and Reactor Coolant Pump Supports - (A-12)
9. Systems Interaction in Nuclear Power Plants - (A-17)
10. Environmental Qualification of Safety-Related Electrical Equipment - (A-24)
11. Reactor Vessel Pressure Transient Protection - (A-26)
12. Residual Heat Removal Requirements - (A-31)
13. Control of Heavy Loads Near Spent Fuel - (A-36)
14. Seismic Design Criteria - (A-40)
15. Pipe Cracks at Boiling Water Reactors - (A-42)
16. Containment Emergency Sump Reliability - (A-43)
17. Station Blackout - (A-44)

Six of the 22 tasks identified as the "Unresolved Safety Issues" are not applicable to WNP-2 because they apply to pressurized water reactors only. These tasks are A-2, A-3, A-4, A-5, A-12, and A-26. Also, Task A-6 and A-7 only apply to Mark I boiling water reactor containments. The NRC staff has issued NUREG reports providing its proposed resolution of the seven of 14 remaining tasks that are applicable to WNP-2. Below is a list of those issues.

Task No.	NUREG Report and Title	Addressed in FSAR Section
A-8	NUREG-0487, "Mark II Containment Lead Plant Program Load Evaluation and Acceptance Criteria", October 1978. Supplement 1 to NUREG-0487, October 1980. Supplement 2 to NUREG-0487, February 1981.	6.2, Appendix G
A-10	NUREG-0619, "BWR Feedwater Nozzle and Control Rod Drive Return Line Nozzle Cracking"	5.2
A-24	NUREG-0588, "Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment"	3.11
A-31	SRP 5.4.7 and BTP 5-1, "Residual Heat Removal Systems" incorporate requirements of USI A-31.	5.4
A-36	NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants".	9.1
A-39	NUREG-0487 and Supplement 1 to NUREG-0487 (see above).	6.2, Appendix G
A-42	NUREG-0313, Revision 1, "Technical Report on Material Selection and Processing Guidelines for BWR Coolant Pressure Boundary Piping".	5.2

The remaining issues applicable to WNP-2 are listed below:

1. Water Hammer (A-1)
2. Anticipated Transients Without Scram (A-9)
3. Reactor Vessel Materials Toughness (A-11)
4. Systems Interaction in Nuclear Power Plants (A-17)
5. Seismic Design Criteria (A-40)
6. Containment Emergency Sump Reliability (A-43)
7. Station Blackout (A-44)

The applicability and bases for licensing prior to ultimate resolution of the above listed Unresolved Safety Issues are discussed in Section 1.5.2.

1.5.1.2 New "Unresolved Safety Issues"

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

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

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

The applicability and bases for licensing prior to ultimate resolution of the four new Unresolved Safety Issues for WNP-2 are also discussed in Section 1.5.2.

1.5.2 DISCUSSION OF UNRESOLVED SAFETY ISSUES AS THEY RELATE TO WNP-2

A-1 <u>Water Hammer</u>

Description:

Water Hammer events are intense pressure pulses in fluid systems caused by any one of a number of mechanisms and system conditions such as rapid condensation of steam pockets, steam-driven slugs of water, pump startup with partially empty lines, and rapid

valve motion. Since 1971 there have been over 200 incidents involving water hammers in BWRs and PWRs reported. The water hammers (or steam hammers) have involved steam generator feed rings and piping, the RHR system, ECCS systems and containment spray, service water, feedwater and steam lines. Most of the damage reported has been relatively minor, involving pipe hangers and restraints; however, there have been several incidents which have resulted in piping and valve damages.

WNP-2 Position:

WNP-2 has installed a system to preclude water hammer from occurring in emergency core cooling system lines. This system consists of water leg pumps to keep the ECCS lines water-filled so that ECCS pumps will not start pumping into voided lines and steam will not collect in the ECCS piping. To ensure that the ECCS lines remain water-filled, vents have been installed and a technical specification requirement to periodically vent air from the lines has been imposed.

Approaches used at 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. WNP-2 has committed to conduct a preoperational vibration dynamic effects program in accordance with Section III of ASME for all 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 water hammer event, core cooling is assured by the emergency core cooling systems and protection against the dynamic effects of such pipe breaks inside and outside of containment is provided.

In the event that Task A-1 identifies potentially significant water hammer scenarios which have not explicitly been accounted for in the design and operation of WNP-2, corrective measures may be required at that time. The task has not identified the need for measures beyond those already implemented.

Based on the foregoing, we conclude that WNP-2 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

Description:

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.

WNP-2 Position:

A recirculation pump trip provision has been incorporated in the WNP-2 design. In addition, the Supply System has implemented emergency procedures and operator training to cope with potential anticipated transient without scram events.

Operator training and action as desired, in conjunction with the automatic recirculation pump trip significantly improve the capability of the facility to withstand a range of anticipated transient without scram events, such that operation of this facility presents no undue risk to the health and safety of the public while this matter is under NRC review. The ATWS issue is currently being reviewed by the Commission. A proposed rule was published in the Federal Register on November 24, 1981. This proposed rule is presently being reviewed by the Supply System. The Supply System will be required to meet the requirements of the final ATWS rule, which is anticipated in mid-to-late 1982.

A-11 Reactor Vessel Material Toughness

Description:

Because the possibility of failure of nuclear reactor pressure vessels designed to the ASME Boiler and Pressure Vessel Code is remote, the design of nuclear facilities does not provide protection against reactor vessel failure. However, as plants accumulate more and more service time, neutron irradiation reduces the material fracture toughness and initial safety margin.

Results from reactor vessel surveillance programs indicate that up to approximately 20 operating nuclear reactors will have beltline materials with marginal toughness, relative to the requirements of Appendix G and H of 10CFR50 after comparatively short periods of operation. For most plants now in licensing process, current criteria, together with the materials currently employed, are adequate to ensure suitable safety margins for reactor vessels throughout their design lives. However, a few plants under licensing reviews have reactor vessels that have been identified as having the potential for marginal fracture toughness within their design lives; these vessels will have to be reevaluated in the light of the new criteria for long-term acceptability.

WNP-2 Position:

The materials used in fabricating the WNP-2 vessel were selected to assure that suitable safety margins will exist over the life of the plant; including the degrading effects of radiation on material toughness. Additionally, WNP-2 reactor will be operated with restrictions imposed by technical specifications on the pressure during heatup and cooldown operations. These restrictions assure that the reactor vessel will not be subjected to a combination of pressure and temperature that could cause brittle fracture of the vessel if there were significant flaws in the vessel material. The effect of neutron radiation on the fracture toughness of the vessel material over the life of the plant is accounted for in technical specification limitations.

Based on the information included in FSAR Section 5.3, we can conclude that WNP-2 will have adequate safety margins against brittle failure during operating, testing, maintenance, and anticipated transient conditions over the life of the plant.

Material surveillance programs (FSAR Section 5.3.1) and Inservice Inspection Programs (FSAR Section 5.2.4) are in accordance with applicable ASEM Code requirements, and provide assurance that brittle fracture control and pressure vessel integrity will be maintained throughout the service lifetime of the reactor pressure vessel.

Based on the foregoing, we conclude that WNP-2 can be operated prior to ultimate resolution of this generic issue without undue risk to the health and safety of the public.

A-17 Systems Interaction in Nuclear Power Plants

Description:

The design and analyses by the plant designers, and the subsequent review and evaluation by the NRC staff take into consideration the interdisciplinary areas of concern and account for systems interaction to a large extent. Furthermore, many of the NRC regulatory criteria are aimed at controlling the risks from systems interactions. Examples include the single failure criterion and separation criteria.

Nevertheless, there is some question regarding the interaction of various plant systems, both as to the supporting roles such systems play and as to the effect one system can have on other systems, particularly with regard to whether actions or consequences could adversely affect the presumed redundancy and independence of safety systems.

The problem to be resolved by this task is to identify where the present design, analysis, and review procedures may not acceptably account for potentially adverse systems interaction and to recommend the regulatory action that should be taken to rectify deficiencies in the procedures. (Also, see TMI Action Plan NUREG-0660, Item II.C.3.)

WNP-2 Position:

WNP-2 has been designed to meet the licensing 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. The design provisions are supplemented by NRC staff review procedures of the Standard Review Plan which require interdisciplinary reviews of potential systems interactions, provide for an adequately safe situation with respect to such interactions. The quality assurance program which is followed during design, construction, and operational phases for WNP-2 is expected to provide added assurance against the potential for adverse systems interactions.

In mid-1977, Task A-17 was initiated by NRC to confirm the 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. The Supply System followed somewhat similar procedures and provided 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 Systems Interaction Branch, formed in the Office of Nuclear Reactor Regulation in April 1980, has been studying state-of-the-art methods that can be used to predict systems interactions. The initial effort, supported by three laboratory contractors, is underway; a range of methods is being considered and tested for feasibility against a sample of some systems interaction candidates derived from Licensee Event Report evaluations.

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

Therefore, we conclude that there is reasonable assurance that WNP-2 can be operated prior to the final resolution of this generic issue without endangering the health and safety of the public.

A-40 <u>Seismic Design Criteria - Short-Term Program</u>

Description:

NRC regulations require that nuclear power plant structures, systems, and components important to safety be designed to withstand the effects of natural phenomena such as earthquakes. Detailed requirements and guidance regarding the seismic design of nuclear plants are provided in the NRC regulations and in regulatory guides issued by the Commission. However, there are a number of plants with construction permits and operating licenses issued before the NRC's current regulations and regulatory guidance were in place. For this reason, re-reviews of the seismic design of various plants are being undertaken to assure that these plants do not present an undue

risk to the public. Task A-40 is, in effect, a compendium of short-term efforts to support such reevaluation efforts of the NRC staff, especially those related to older operating plants. In addition, some revisions to sections of the Standard Review Plan and regulatory guides to bring them more in line with the state-of-the-art will result.

WNP-2 Position:

WNP-2 plant structures, systems, and components important to safety are designed to withstand the effects of natural phenomena such as earthquakes using current licensing criteria and requirements.

The seismic design basis and seismic design of WNP-2 are detailed in FSAR Sections 3.7 and 3.8. Should the resolution of Task A-40 indicate a change is needed in these licensing requirements, WNP-2 design will be evaluated accordingly. Therefore, we conclude WNP-2 can be operated prior to ultimate resolution of this issue without endangering the health and safety of the public.

A-43 <u>Containment Emergency Sump Performance</u>

Description:

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

Concern addressed in this issue for boiling water reactors is limited to the potential for degraded emergency core cooling system performance as a result of thermal insulation debris that may be blown into the suppression pool during a loss-of-coolant accident and cause blockage of the pump suction lines.

WNP-2 Position:

The blown off insulation panels constitute the only credible debris within the primary containment following a LOCA and seismic event. Large pieces of debris are not considered to

have deleterious effects on the containment systems. The grating at the 501'-0" elevation, which covers approximately 80% of the primary containment cross-sectional area would stop the majority of the loose insulation panels. The potential debris in the drywell could only be swept into the suppression pool via the downcomer piping. However, the downcomer pipes (approximately two feet in diameter) are capped with jet deflectors and would prevent large pieces from reaching the suppression pool. In addition, each ECCS pump suppression pool suction consists of a pipe 'T' with a suction screen assembly at each end. Accordingly, we conclude that WNP-2 can be operated prior to ultimate resolution of this generic issue without endangering the health and safety of the public.

A-44 Station Blackout

Description:

The unlikely, but possible, loss of all AC power (that is, the loss of AC power from the offsite source and from the onsite source) is referred to as a station blackout. In the event of a station blackout, the capability to cool the reactor core would be dependent on the availability of systems which do not require AC power supplies, and on the ability to restore AC power in a timely manner. The concern is that the occurrence of a station blackout may be a relatively high probability event and that the consequences of this event may be unacceptable, for example, severe core damage may result.

Review your plant operations to determine your capability to mitigate a station blackout event and promptly implement as necessary, emergency procedures and a training program for station blackout events (for details see NRC Generic Letter 81-04, dated February 25, 1981).

WNP-2 Position:

The loss of all alternating current power was not a design basis event for WNP-2. Nonetheless, a combination of design, operating, and testing requirements implemented by the Supply System will assure that WNP-2 will have substantial resistance to a loss of all AC power and that, even if a loss of all AC power should occur, there is reasonable assurance that the core will be cooled. These are described below.

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A loss of offsite AC power involves a loss of both preferred and backup sources of offsite power. Loss of all offsite power for WNP-2 is a relatively unlikely event. The plant is tied into the Bonneville Power Grid which is based mostly on hydroelectric sources and is considered one of the most reliable major distribution systems, based on actual data. The design, inspection, and testing provisions for the offsite power system are described in Section 8.2 of the FSAR. Data cited in a recent report (Reference 1.5-1) indicate that the mean time between failures for simultaneous loss of both independent lines in the general location of WNP-2 is about 30 years for outages between two-second and twenty-minute duration, and it is over 110 years for outages over twenty-minute duration. These figures are based on Bonneville Power Administration records.

If offsite AC is lost, three diesel generators (two emergency and one HPCS) and their associated distribution system will deliver emergency power to safety-related equipment. The design, testing, surveillance, and maintenance provisions for onsite emergency diesels are described in Section 8.3 of the FSAR.

If both offsite and onsite AC power are lost, WNP-2 may use a combination of safety/relief valves and reactor core isolation system to remove core decay heat without reliance on AC power. These systems assure that adequate cooling can be maintained for at least two hours which allows time for restoration of AC power from either offsite or onsite sources.

The issue of station blackout was also considered by the Atomic Safety and Licensing Board (ALAB-603) for the St. Lucie No. 2 facility. In addition, in view of the completion schedule for Task A-44, 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. Consequently, NRC requested (Generic letter 81-04, dated February 25, 1981) a review of plant operation to determine the Supply System's capability to mitigate a station blackout event and promptly implement, as necessary, emergency procedures and training programs for station blackout events.

Appropriate review of procedures and training programs for station blackout events will be completed prior to fuel load date.

Based on the above, we conclude that there is reasonable assurance that WNP-2 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

Description:

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

WNP-2 Position:

The WNP-2 reactor has various methods for the removal of decay heat. After a reactor trip, three "regular" modes of operation at high system pressure are available for heat removal and coolant makeup:

1. If turbine bypass, main condenser and feedwater pumps are available, one of the feedpumps is used for coolant makeup. Steam is released through the four main steam lines and the turbine bypass into the main condenser. The turbine bypass opens periodically on a pressure signal such that no relief valve actuation is required.

2. If any of the above needed equipment is not available, or if containment isolation has occurred, the HPCS (high pressure core spray) system together with the RCIC (reactor core isolation cooling) system is used for about 25 minutes, after which the RCIC system alone is sufficient. In case the HPCS system is not available, the RCIC system can be used alone. In that case, some level decrease in the reactor vessel will occur during the first 25 minutes, and some steam will be released through the relief valves during that time. The reactor will operate at a saturation

pressure of 1,076 psig, which is the setpoint of the two lowest pressure relief valves. Condensation of the released steam will result in a modest heatup of the suppression pool.

3. If it is desired to avoid reactor cooldown (maintain hot shutdown conditions), the RCIC is used together with the RHR (residual heat removal) system in the steam condensing mode. Reactor steam is routed through pressure reducing valves to the RHR heat exchangers where it is condensed. The condensate, together with the RCIC turbine condensate, is pumped back into the reactor by the RCIC injection pump.

Several variations of these operational modes can be envisioned. One additional source of high pressure coolant are the control rod drive pumps. On older reactor designs, which have a steam driven HPCI (high pressure collant injection) system instead of an electrically driven HPCS system, there have been cases where the HPCI system was not immediately available on demand. The RCIC system then was used alone, and in all cases known to us, the HPCI system was brought on line within about 30 minutes.

In the unlikely case that none of the above mentioned equipment can be operated, manual or automatic depressurization of the reactor is available. Depressurization will decrease the vessel pressure to below 300 psig. where LPCS (low pressure core spray) and LPCI (low pressure coolant injection) can be used. The RCIC and the HPCS at WNP-2 have improvements over comparable systems at older boiling water reactors. The reactor core isolation cooling system has been upgraded to safety-grade quality (now required for all boiling water reactors), and the high pressure core spray is powered by its own dedicated diesel so it can operate with an assumed loss of all other sources of alternating current power. Also, the residual heat removal system contains three pumps; the flow capacity of any single pump is sufficient to easily remove the decay heat. Accordingly, we conclude that WNP-2 can be operated prior to ultimate resolution of this generic issue without endangering the health and safety of the public.

A-46 <u>Seismic Qualification of Equipment in Operating Plants</u>

Description

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 funtion is not required for safe shutdown, but whose failure could result in adverse conditions which might impair shutdown functions.

WNP-2 Position:

WNP-2 electrical and mechanical equipment are designed using seismic criteria delineated in IEEE 344-1971 and the design is being reviewed and approved by the NRC staff using current design criteria and methods for seismic qualification.

A-47 Safety Implications of Control Systems

Description:

This issue concerns the potential for accidents or transients 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 and would be in addition to any control system failure that may have initiated the event. Although it is generally believed that control system failures are not likely to result in loss of safety functions which could lead to serious events or result in conditions that safety systems are not able to cope with, in-depth studies have not been performed to support this belief. The potential for an accident that would affect a particular control system -- and the effects of the control system failures -- will differ from plant to plant. Therefore, it is not likely that it will be possible to develop generic answers to these concerns, but rather plant-specific reviews will be required. The purpose of this Unresolved Safety Issue is to define generic criteria that may be used for plant-specific reviews. A specific subtask of this issue will be to study the steam generator overfill transient in PWRs and the reactor overfill transient in BWRs to determine and define the need for preventive and/or mitigating design measures to accommodate this transient.

WNP-2 Position:

The WNP-2 control and safety systems have been designed with the goal of ensuring that control system failures (either single or multiple failures) will not prevent automatic or manual initiation and operation of any safety system equipment required to trip the plant or to maintain the plant in a safe shutdown condition following any "anticipated operational occurrence" or "accident". This has been accomplished by either providing independence between safety and nonsafety systems or providing isolating devices between safety and nonsafety systems. These devices preclude the propagation of nonsafety system equipment faults such that operation of the safety system equipment is not impaired.

A wide range of bounding transients and accidents is presently analyzed to assure that the postulated events would be adequately mitigated by the safety systems. In addition, in response to NRC Questions 031.135, 031.137, and 031.138, systematic reviews of safety systems is being performed with the goal of ensuring that the control system failures (single or multiple) will not defeat safety system action. Specifically, these reviews will include:

1. IE Bulletin 79-27 (Question 031.135)

   A series of tables will be developed which lists power sources to include alarm indications, instruments and control devices on these power sources. The primary and secondary effects on safe shutdown from loss of the power sources to each load will be analyzed. Design and/or procedure modification will be made as necessary when the determined effects have an adverse impact on plant safety.

2. Control System Failure (Question 031.138)

   The review procedure being followed to address this question is:

   a. Define bus structure throughout the plant from the grid to the lower voltage level.
   b. Identify loads.
   c. Eliminate loads where failure or malfunction would not impact plant safety.
   d. Identify instruments (related to the above non-eliminated loads) on common instrument taps/impulse lines/hydraulics.
   e. List effects of failure or malfunction of each load/common sensors due to failure or malfunction of power source/plugged or broken lines.

f. Analyze the effects on systems important to plant safety.
g. Analyze the combined effects due to plugged or broken lines/cascading power losses.
h. Compare results of these analyses to those already covered in Chapter 15.
i. Modify and/or augment Chapter 15 as necessary such that failures or malfunctions of common power source or sensor would not require action or response beyond the capability of operations or safety systems.

3. IE Information Notice 79-22 (Question 031.137)

A matrix will be developed to indicate the effects of non-safety grade/control equipment, subjected to the adverse environment of a high energy line break, or the protection functions performed by the safety grade equipment. If interaction is discovered then the impact of failure of the applicable system upon the safety analyses will be evaluated.

Several early boiling water reactors have experienced reactor vessel overfill transients with subsequent two-phase or liquid flow through the safety/relief valves. Following these early events, high level trips (level 8) have been installed at WNP-2 and other operating BWRs to terminate flow from the appropriate systems. These high-level trips are single failure proof and periodic surveillance is required by the Technical Specifications. No overfilling events have been reported at this or other operating BWRs since the level 8 trips were installed.

Based on the above, the Supply System concludes that there is a reasonable assurance that WNP-2 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

Description:

Postulated reactor accidents which result in a degraded or melted core can result in generation and release to the containment of large quantities of hydrogen. The hydrogen is formed from the reaction of the zirconium fuel cladding with steam at high temperatures and/or by radiolysis of water. Experience gained from the TMI-2 accident indicates that we may want to require more specific

design provisions for handling larger hydrogen releases than currently required by the regulations particularly for smaller, low pressure containment designs.

This issue will investigate means to predict the quantity and release rate of hydrogen following degraded core accidents and various means to cope with large releases to the containment such as inerting of the containment or controlled burning. The potential effects of proposed hydrogen control measures on safety including the effects of hydrogen burnes on safety related equipment will also be investigate.

WNP-2 Position:

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

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

WNP-2 has committed to inerting the containment buildings during power operation. The Supply System concludes that WNP-2 can be operated prior to resolution of this unresolved safety issue and the proposed rulemaking without undue risk to the health and safety of the public.

1.5 References:

1.5-1 WPPSS-ENT-087, Reliability Analysis of the Auxiliary Heat Removal Systems for the Washington Nuclear Project Number 2, April 1981, Washington Public Power, Richland, Washington.