PUGET POWER

February 22, 1982
PLN-245

Mr. Harold R. Denton, Director
Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Subject: Puget Sound Power & Light Company
Skagit/Hanford Nuclear Project, Units 1 & 2
Docket Nos. 50-522 and 50-523
Request for Information on Unresolved
Safety Issues

Reference: Robert L. Tedesco letter to F. Spangenberg
dated January 25, 1982

Dear Mr. Denton:

Attached are our responses to the above referenced letter concerning generic unresolved and technically resolved safety issues applicable to the Skagit/Hanford Nuclear Project.

Very truly yours,

Robert V Myers

Robert V. Myers
Vice President
Generation Resources

Attachment

cc: R. Tedesco, NRC
E. Adensam, NRC
M. Mallory, NRC

B001
S 1/1

S/HNP RESPONSES TO GENERIC UNRESOLVED AND TECHNICALLY RESOLVED SAFETY ISSUES

A-1 Waterhammer

Description

Waterhammer events are intense pressure pulses in fluid systems caused by any one of a number of mechanisms and system conditions such as condensation of steam pockets, steam-driven slugs of water, pump startup with partially empty lines, and rapid valve motion. Since 1971 over 200 incidents involving waterhammer in pressurized and boiling water reactors have been reported. The waterhammers (or steam hammers) have involved steam generator feedrings and piping, the residual heat removal systems, emergency core cooling systems, and containment spray, service water, feedwater and steam lines.

Most of the damage reported has been relatively minor, involving pipe hangers and restraints; however, several waterhammer incidents have resulted in piping and valve damage. The most serious waterhammer events have occurred in the steam generator feedrings of pressurized water reactors. In no case has any waterhammer incident resulted in the release of radioactive material.

Response

Although waterhammer can occur in any light water reactor and over 100 actual and probable events have been reported in boiling water reactors, none have caused major pipe failure in a boiling water reactor such as S/HNP's and none resulted in the offsite release of radioactivity. As noted above, the most severe waterhammers observed to date have been in steam generators. Since the boiling water reactor does not utilize a steam generator, these worst cases are eliminated. Furthermore, any waterhammer which may occur in feedwater or main steam piping will not impair the emergency core cooling system since all ECCS water enters the reactor vessel via five separate reactor vessel nozzles independent of the feedwater and main steam piping.

In order to protect the emergency core cooling system against the effects of waterhammer, each ECCS pump is provided with its own jockey pump which provides a continuous supply of water to the emergency core cooling system discharge piping. Further assurance for filled discharge piping is provided by pressure instrumentation. An alarm sounds in the main control room if the pressure falls below

A-1 Waterhammer (con't)

a predetermined setpoint indicating difficulty in maintaining a filled discharge line. To ensure that the emergency core cooling system lines remain water-filled, vents will be installed and a Technical Specification requirement to periodically vent air from the lines will be imposed.

With regard to additional protection against potential waterhammer events currently provided in plants, piping design codes require consideration of impact loads. Approaches used at the design stage include: (1) increasing valve closure times, (2) piping layout to preclude water slugs in steam lines and vapor formation in water lines, (3) use of snubbers and pipe hangers, and (4) use of vents and drains.

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

In the event that potentially significant waterhammer scenarios are identified which have not explicitly been accounted for in the design and operation of S/HNP, corrective measures will be implemented at that time. The need for measures beyond those already implemented has not been identified.

Based on the foregoing we conclude that the S/HNP can be constructed prior to ultimate resolution of this generic issue without undue risk to the health and safety of the public.

A-1 Waterhammer (con't)

a predetermined setpoint indicating difficulty in maintaining a filled discharge line. To ensure that the emergency core cooling system lines remain water-filled, vents will be installed and a Technical Specification requirement to periodically vent air from the lines will be imposed.

With regard to additional protection against potential waterhammer events currently provided in plants, piping design codes require consideration of impact loads. Approaches used at the design stage include: (1) increasing valve closure times, (2) piping layout to preclude water slugs in steam lines and vapor formation in water lines, (3) use of snubbers and pipe hangers, and (4) use of vents and drains.

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

In the event that potentially significant waterhammer scenarios are identified which have not explicitly been accounted for in the design and operation of S/HNP, corrective measures will be implemented at that time. The need for measures beyond those already implemented has not been identified.

Based on the foregoing we conclude that the S/HNP can be constructed 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 systems 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.

Resolution

As noted in Federal Register Vol. 46, No. 226, 57521 to 57532, Nuclear Regulatory Commission 10 CFR 50 "Standards for the Reduction of Risk from Anticipated Transients Without Scram (ATWS) Events for Light-Water-Cooled Nuclear Power Plants", the Commission is considering three alternate proposed rules. These proposed rules are mutually exclusive in that there will be only one final rule. The comment period for each of the proposed rules expires April 23, 1982, indicating that issuance of the final rule will be made in a timely manner for incorporation into the S/HNP design. Implementation of the potential ATWS requirements defined by any of the three proposed rules will not be compromised by existing S/HNP design or design work performed prior to the final rule.

S/HNP design details demonstrating compliance with the final rule will be incorporated in the Final Safety Analysis Report.

A-10 BWR Feedwater Nozzle and Control Rod Drive Return Line Nozzle

Description

Generic Technical Activity (GTA) A-10 is concerned with cracking found in feedwater nozzles at several operating BWRs. The cracks have been discovered in the nozzle blend radius and bore region. The crack growth is slow but accelerates with increasing depth. It is possible that the cracks will present a repair problem if ASME code limits for nozzle reinforcement are exceeded during crack removal by grinding. Similar cracking has also been discovered on BWR control rod drive (CRD) return line nozzles.

Resolution

Feedwater Nozzle Cracking -

Issuance of NUREG-0619 resolves Generic Technical Activity A-10. The NRC staff concluded that the GE triple thermal sleeve feedwater sparger modification, when combined with the removal of stainless steel cladding, reroute of the reactor water cleanup system to the feedwater system ahead of the nozzles, and appropriate operating procedures will provide a substantial and acceptable improvement over previous designs. Skagit/Hanford will utilize the GE triple thermal-sleeve feedwater sparger as described in NEDE-21821-A. The Skagit/Hanford reactor pressure vessel is not clad in the nozzle area. The Skagit/Hanford reactor water cleanup (RWCU) system is routed to the feedwater system (see 251 GESSAR, Figure 5.5-12b and S/HNP PSAR Figure 10.4-5).

According to the NRC Safety Evaluation Report, included as Appendix C to NUREG-0619, the triple thermal-sleeve sparger design may be used without further justification beyond that given by GE. As for ultrasonic testing and inspections, the Staff conclusions in Sections 6 and 7 of Appendix C will be used as guidance in developing inspection and testing procedures. New testing techniques will be examined for applicability as they are developed.

Control Rod Drive Return Line Nozzle -

The control rod drive (CRD) return line will be deleted and the CRD return line nozzle will be capped on the Skagit/Hanford pressure vessel, as permitted by Part II, Section 8.1 (4) of NUREG-0619. This design change information will be provided in the Skagit/Hanford FSAR.

A-11 Reactor Vessel Materials Toughness

Description

Because the possibility of failure of nuclear reactor pressure vessels (RPV) 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 margins. The reduction in fracture toughness is accelerated with increased copper and phosphorous content in the RPV material.

Resolution

To assure adequate safety margins, adjustment to the nil ductility transition temperature (NDTT) and the developmental method for pressure/temperature curves are specified in 10 CFR 50 Appendices G and H. The amount of adjustment to the operating curves is a function of reference temperature, RT_{NDT} which depends upon the fast neutron (½1 MEV) fluence and copper and phosphorous content in the RPV material. For BWR/6s, the copper and phosphorous content of the material is closely controlled. Furthermore, high upper shelf toughness is specified. The fast neutron fluence in BWRs is relatively low because of the additional moderator (water) between the core and the RPV wall. Therefore the reactor pressure vessel material toughness issue is of relatively low concern for BWR/6s.

For the Skagit/Hanford pressure vessel the limiting material in the core belt line will be determined and the initial RT_{NDT} will be established. Based on a predicted adjusted reference temperature as a function of fluence and copper and phosphorous content, the end-of-life RT_{NDT} will be determined. By this means the Skagit/Hanford reactor pressure vessel will be shown to have an adequate safety margin with respect to the requirements of 10 CFR 50 Appendices G and H. Values for the material content are within the following values specified for the reactor pressure vessel: copper - 0.08%; phosphorous (plate) - 0.015% and phosphorous (welded material) - 0.20%. The initial and end-of-life RT_{NDT} will be provided at the FSAR stage.

A-17 System Interaction in Nuclear Power Plants

Description

The licensing requirements and procedures used in the design address many different types of systems interaction. Current licensing requirements are founded on the defense-in-depth principle. Adherence to this principle results in requirements such as physical separation and independence of redundant safety systems, and protection against events such as high energy line ruptures, missiles, high winds, flooding, seismic events, fires, operator errors, and sabotage. These design provisions supplemented by the current review procedures of the Standard Review Plan (NUREG-0800), 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.

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.

Response

The project administrative procedures provide the required guidance for interface between the applicant, GE, Bechtel and vendors. Specifically, the project procedures manuals identify the division of responsibility between the applicant, GE (NSSS supplier), Westinghouse (Turbine Generator supplier) and Bechtel. These responsibilities consist of establishing Basic Design Data, Detailed Design, Design Review, Procurement, Construction Management, and Start-up Procedures.

The Bechtel Project Engineering Procedures Manual identifies Bechtel's design interface requirements. These requirements control internal, external and interdiscipline design review processes which include interface between Bechtel and the applicant, GE, Westinghouse, suppliers/subcontractors and

A-17 Systems Interaction in Nuclear Power Plants (con't)

consultants. The Procedures Manual contains provisions with regard to communications, documentation and change control. The Project Engineering Team also interfaces with the following Bechtel entities: Home Office Construction Department, Startup Engineering, Specialist Groups; e.g., the Chiefs of Engineering and their staffs, other divisions and companies; e.g., Hydro and Community Facilities (H&CF) and Research & Engineering (R&E).

The design control review and verification procedures used by the applicant, and described in PSAR subsections 17.1.3 through 17.1.16 are the general methods by which the concerns identified in Task A-17 are mitigated. Further to this general approach, the joint interdisciplinary system design reviews conducted between the applicant and Bechtel/GE, in which all engineering documents and the S/HNP engineering model are addressed and cross checked, are the specific methods by which potentially adverse systems interactions are noted and rectified. Additional details of the joint inter-disciplinary relationship are presented in the S/HNP PSAR Amendment 23, Appendix 1B (I.F.2 and II.J.3.1).

General Electric has been approaching the issue with the use of thorough interdisciplinary reviews. This process makes use of design reviews, design verifications and audits to bring any potential systems interaction to light. In addition, General Electric is represented in the AIF Systems Interaction Subcommittee to keep abreast of current NRC approaches on the issue. This helps us to address specific issues as soon as they arise.

In addition, the interface between Bechtel, General Electric, and the utility is tracked by the Communication Register/Action List. A control number from the Register is assigned to any correspondence that requires action by the recipient. This control number enables the item to be tracked and ensures a follow-up on any open item for which a response has not been received.

The Project will conduct in-plant walkdowns near the end of construction to provide additional assurance that the as-built condition of the plant does not contain obvious potential hazards to safety-related equipment.

The following safety issues are included in the walkdown program:

A-17 Systems Interaction in Nuclear Power Plants (con't)

Seismic Category II over Seismic Category I
High Energy Line Break
Flooding
Jet Impingement

Power and control cables are separated into three independent electrical divisons -- 1, 2, and 3 -- each serving separate safety-related systems. Operation of Division 1 only or operation of Division 2 only is sufficient to achieve safe shutdown. The operability of Division 1 or 2 is assured by fire protection measures taken to ensure that a single fire cannot disable both divisions. Separation criteria utilized during the installation of safety-related cables provide protection against disabling redundant safety-related equipment by a cable fire. The criteria used for separation of safety-related cable trays and conduits are based on Regulatory Guide 1.75. The intent is to prevent a possible fire in one safety-related cable tray from spreading into a safety-related cable tray of a redundant electrical division and to prevent a possible fire in a non-safety-related cable tray from spreading into any safety-related tray.

Adverse systems interactions due to human errors will be minimized by applying human factors engineering principles to the control room design as described in the response to NUREG-0718 Item I.D.1 (PSAR Appendix 1B).

A-24 Environmental Qualification of Safety-Related Electrical Equipment

Description

Several aspects of equipment qualification are being pursued by the NRC staff and the nuclear industry on a generic basis, in order to achieve a more uniform implementation of requirements established in IEEE Standard 323-1974. Generic Task A-24 involves the development of the NRC's interim staff position regarding how the requirements of IEEE Standard 323-1974 can be met.

Resolution

As presented in Sections 3.11, 7.1 and 8.3 of the S/HNP PSAR and in Sections 3.11 and 7.1 of the 251 NSSS GESSAR, which is incorporated as part of the S/HNP PSAR, S/HNP Class 1E equipment qualification is based on IEEE 323-1974. The commitments and outlined program described in the aforementioned sections were found to be acceptable by the NRC in Section 3.11 of the Skagit SER, NUREG-0309, September 1977.

The NRC has issued a proposed rule, "Environmental Qualification of Electric Equipment for Nuclear Power Plants", in the Federal Register on January 20, 1982 at pages 2,876 - 2,879. The comment period for the proposed rule expires March 22, 1982 indicating that the final rule can be anticipated to be published in a timely manner for incorporation into the S/HNP's equipment qualification efforts.

The details of S/HNP's conformance to the final rule will be presented in the appropriate sections of the S/HNP Final Safety Analysis Report. Such a program for the detailed equipment qualification was also found acceptable by the NRC in Section 3.11 of the Skagit SER, NUREG-0309, September 1977.

A-31 Residual Heat Removal Requirements

Description

Task A-31 investigated the ability of the Residual Heat Removal (RHR) system to adequately bring the plant to a cold shutdown condition. The revised Standard Review Plant (SRP) Section 5.4.7 and BTP 5-1 require compliance with GDC 19 and 34.

Resolution

Compliance of Skagit/Hanford to the requirements of GDC 19 and 34 is documented in Subsections 3.1.2.4.5, 5.5.7 and 15.1.27 of the 251 NSSS GESSAR, and Subsection 3.1.2.2.10.1 of the Skagit/Hanford Nuclear Project PSAR. This compliance was found acceptable by the NRC in Section 5.4.5 of the September 1977 SER (NUREG-0309) and the October 1978 SER Supplement 1 (NUREG-0309, Supplement 1) for the Skagit Nuclear Power Project.

A-36 Control of Heavy Loads Near Spent Fuel

Description

Overhead handling systems (cranes) are used to lift heavy objects in the vicinity of spent fuel in PWRs and BWRs. If a heavy object, e.g., a spent fuel shipping cask or shielding block, were to fall or tip on to spent fuel in the storage pool or the reactor core during refueling and damage the fuel, there would be a release of radioactivity to the environment and a potential for radiation over-exposure to inplant personnel. If the dropped object is large, and the damaged fuel contained a large amount of undecayed fission products, radiation releases to the environment could exceed 10 CFR Part 100 guidelines. These requirements are currently considered in the licensing review. However, with the advent of increased and longer term storage of spent fuel assemblies in the spent fuel pools, there is a need to systematically review NRC requirements, facility designs, and technical specifications requiring the movement of heavy loads to assess safety margins and to improve those margins where warranted.

Resolution

The S/HNP spent fuel storage, spent fuel cask handling, and fuel handling systems are described in Section 9.1 of the PSAR. These systems were reviewed by the NRC staff and judged acceptable with respect to the control of heavy loads near spent fuel, as documented in Section 9.1 of the Skagit SER (NUREG-0309, September 1977).

A-39 SRV Pool Dynamic Loads

Description

BWR plants are equipped with relief valves that discharge into the suppression pool. Upon relief valve actuation, the initial air column within the SRV discharge line is accelerated by the high pressure steam flow and expands as it is released into the pool as a high pressure air bubble. The high rate of air and steam injection flow in the pool followed by expansion and contraction of the bubble as it rises to the pool surface produces pressure oscillations in the pool boundary. The effect is referred to as the air-cleaning phenomenon.

Following the air-clearing phase, pure steam is injected into the pool. Condensation oscillations occur during this time period. However, the amplitudes of these vibrations are relatively small at low pool temperatures. Continued blowdown into the pool will increase the pool temperature until a threshold temperature is reached. At this point, steam condensation becomes unstable. Vibrations and forces can increase by a factor of 10 or more if the SRV continues to blow down. This effect is referred to as the steam quenching vibration phenomenon. Current practice for BWR operating plants is to restrict the allowable operating temperature envelope via Technical Specifications such that the threshold temperature is not reached.

Resolution

The generic review of Mark III SRV loads is being conducted on the GESSAR II document and is applicable to S/HNP. The final phase of NRC review of the Mark III SRV loads began in May 1981 when the NRC staff issued two questions to GE on the reduced SRV load magnitudes (May 21, 1981 letter from K. Kniel - NRC to General Electric, "SRV Pool Dynamic Loads"). An informal meeting was held with the NRC staff in early July, 1981 where GE presented responses to the two questions with a favorable response from the NRC staff. Formal GE responses to the two questions were submitted to the NRC staff in September 1981 (September 13, 1981 letter from J. N. Fox - GE to N. Su - NRC, "SRV Pool Dynamic Loads"). Based on these responses, the NRC staff is expected to accept the GESSAR-II SRV loads and load methodology.

A-40 Seismic Design Criteria

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 guidances were in place. For this reason, 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.

Resolution

The seismic design review of S/HNP has been conducted using current licensing criteria and requirements. The S/HNP seismic design criteria are described in PSAR Section 3.7 and were found acceptable by the NRC in Section 3.7 and 3.8 of the Skagit SER (NUREG-0309, September 1977). The seismic design criteria have been retained in relocating the S/HNP to the Hanford Reservation. It is expected that the criteria will again be found acceptable.

A-42 Pipe Cracks in Boiling Water Reactors

Description

Pipe cracking has occurred in the heat-affected zones of welds in primary system piping in boiling water reactors since the mid-1960s. These cracks have occurred mainly in Type 304 stainless steel that is being used in most operating BWRs. The major cause of this problem has been determined to be intergranular stress corrosion cracking (IGSCC) of austenitic stainless steel components. These components have been made susceptible to this failure mode by being "sensitized" in the narrow heat-affected zone during the welding process.

Resolution

NUREG-0313, Rev. 1, "Technical Report on Material Selection and Processing Guidelines for BWR Coolant Pressure Boundary Piping," addresses Generic Technical Activity A-42 and sets forth acceptable methods to reduce the intergranular stress corrosion cracking susceptability of ASME Code Class 1, 2 and 3 pressure boundary piping and safe ends.

S/HNP will comply with the requirements of NUREG-0313, Rev. 1, and provide a description of such compliance in the FSAR.

A-43 Containment Emergency Sump Reliability

Description

Following a Loss of Coolant Accident (LOCA) in a PWR, water flowing from the break in a primary system would collect on the floor of the containment. During the injection mode, water for core cooling and containment spray is drawn from a large supply tank. When the water reaches a low level in the tank, pumps are realigned to draw from the containment. This is called the recirculation mode wherein water is drawn from the containment floor or sump and pumped to the primary system or containment spray headers. This program addresses the safety issue of adequate sump or suppression pool function in the recirculation mode. It is the objective of this program to develop improved criteria for design, testing, and evaluation which will provide better assurance that emergency sumps will function to satisfy system requirements.

The principal concerns are somewhat interrelated but are best discussed separately. One deals with the various kinds of insulation used on piping and components inside of containment. The concern being that break-initiated debris from the insulation could cause blockage of the sump or otherwise adversely affect the operation of the pumps, spray nozzles, and valves of the safety systems.

The second concern deals with the hydraulic performance of the sump as related to the hydraulic performance of safety systems supplied therefrom. Preoperational tests have been performed on a number of plants to demonstrate operability in the recirculation mode. Adverse flow conditions have been encountered requiring design and procedural modifications to eliminate them. These conditions: air entrainment, cavitation, and vortex formation, are aggrevated by blockage. If not avoided or suppressed, they could result in pump failure during the long term cooling phase following a LOCA.

The concerns relative to debris, blockage and hydraulic performance also apply to boiling water reactors during recirculation from the suppression pools.

Resolution

The S/HNP design for preventing blockage of the ECCS suction lines by insulation debris has been reviewed by the NRC and found acceptable in Section 6.3 of the Skagit SER (NUREG-0309, September 1977).

A-43 Containment Emergency Sump Reliability (con't)

Vortex formation is not considered a problem for S/HNP because approach velocities are low to prevent plugging (as noted in the above SER section) and the suction strainers are located a minimum of 9 feet below the suppression pool surface.

A-44 Station Blackout

Description

Electric power for safety systems at nuclear power plants is supplied by redundant and independent divisions. Each of these electrical divisions includes two offsite alternating current (ac) sources, one onsite ac source (usually diesel-generator), and a direct current (dc) source. Appendix A to 10 CFR 50 defines a total loss of offsite power as an anticipated occurrence, and it is required that independent onsite power supplies be provided at nuclear power plants.

The unlikely but possible loss of ac power (that is, the loss of ac power from the offsite sources 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 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 relativly high probability event and that the consequences of this event may be unacceptable, for example, severe core damage may result.

Resolution

The design of the S/HNP electrical system assures that there will be a source of electrical power for safe shutdown of the reactor. Should there be a loss of both offsite and onsite ac power the plant may use a combination of safety/relief valves and the RCIC system to remove the decay heat without reliance on ac power for not less than two hours. This allows time for restoration of ac power from either offsite or onsite sources.

The loss of offsite ac power involves a loss of two preferred power sources to each division ESF bus. The two physically independent circuits connecting the preferred power sources to the Division 1, Division 2, and Division 3 ESF buses of each unit are from the 500 kV system through the Plant Substation Transformers (PSX) and the Preferred Offsite Transformers (POX). (PSAR Sections 8.1, 8.2 and 8.3).

If all offsite ac power is lost, three diesel generators and their associated distribution systems will deliver power to the safety related equipment. The preferred power sources are continuously monitored at each 4.16 kV bus by voltage relays to detect loss of offsite power or degradation below an acceptable level.

A-44 Station Blackout (con't)

The diesel generators are automatically started on loss of both preferred offsite sources or on LOCA conditions. During non-LOCA conditions, if the normal preferred source is lost and the alternate preferred source is proven to be available and acceptable the ESF bus will be automatically connected (dead bus transfer) to the alternate preferred source approximately two seconds after the undervoltage condition is detected.

During LOCA conditions, if the normal preferred source is lost, the ESF bus will be automatically connected (dead bus transfer) to the diesel generator, when it reaches rated speed and voltage. If required, and if the alternate preferred source is proven to be available and acceptable, the ESF bus may be manually transferred from the diesel generator to the alternate preferred source. The standby power source for each ESF bus is the diesel generator serving exclusively that bus. There is one independent and separate diesel generator for each of the three ESF divisions (PSAR Section 8.3).

The Class 1E DC system is comprised of four independent (Division 1 to 4) 125 volt dc systems. Each division is physically separated to assure that no single credible event will prevent the operation of the required number of redundant functions. The function of the Class 1E DC System is to furnish highly reliable 125 volt dc power for control of power loads, and for instrumentation of equipment that limits the release of fission products and maintains safe plant conditions. Each division of the DC System includes a battery and two battery chargers. One of the two battery chargers is an installed spare. Each battery charger is fed from the ESF bus of the same division (PSAR Section 8.3).

Maintenance and testing programs will be implemented in accordance with detailed design and individual equipment qualification test results. The design accommodates these programs to assure the readiness of these systems to deliver the performance required. (PSAR Sections 8.3.1 and 8.3.2).

A-45 Shutdown Decay Heat Removal Requirements

Description

Task A-45 will investigate the need for possible design requirements to improve the reliability of decay heat removal systems for light water reactors.

Resolution

From the task description in NUREG-0606, this item is primarily concerned with Auxiliary Feedwater Systems for PWRs. However, it also addresses decay heat removal systems for BWRs.

The S/HNP reactors have various methods for removing decay heat. The normal method of removing decay heat is through the steam lines to the main condenser via the turbine bypass system. The condensate is then returned to the reactor through the condensate/feedwater system. If the condensate/feedwater system is not available, water can be provided from condensate storage tank or suppression pool to the reactor vessel through the use of the Reactor Core Isolation Cooling (RCIC) or High Pressure Core Spray (HPCS) systems.

If the condenser is not available, heat can be removed from the reactor by two methods without depressurizing the reactor. First, heat can be removed through the safety-relief valves (SRV) which discharge to the suppression pool. The reactor coolant discharged can then be returned to the reactor vessel via either the RCIC or HPCS systems. Second, heat can be removed by using the Residual Heat Removal (RHR) system in the steam condensing mode operating in conjunction with the RCIC system.

Finally, if both the RCIC and HPCS system are unavailable, the reactor can be depressurized by use of the Automatic Depressurization System (ADS) through the SRVs. Reactor coolant inventory is then maintained by the lower pressure RHR system or Low Pressure Core Spray (LPCS) system. In all of the above modes, heat rejected to the suppression pool is removed by operation of the RHR system in the suppression pool cooling mode.

The RHR System is of safety-grade quality. The RCIC system pump-turbine is steam driven and other system control components are powered from safety related on-site DC power to further enhance system availability. The HPCS system is powered by a dedicated on-site safety-grade diesel generator. The RHR system consists of three pumps, any one

A-45 Shutdown Decay Heat Removal Requirements (con't)

of which can maintain reactor water level. The RHR system and LPCS system can be powered by the redundant on-site safety-grade diesel generators.

In addition, the S/HNP PSAR Appendix 1B item II.B.8(1) in response to NUREG-0718 has committed to a plant/site specific probabilistic risk assessment, one aim of which is to seek improvements in the reliability of core and containment heat removal systems as are significant and practical and do not impact excessively on the plant.

A-46 Seismic Qualification Of Equipment In Operating Plants

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

Resolution

Since the purpose of this item is to address seismic qualification at operating plants, it is of minimum applicability to new plants, such as Skagit/Hanford, which are designed to the current seismic criteria. The seismic design and qualification requirements of mechanical and electrical equipment to be used at Skagit/Hanford are in accordance with the current requirements as discussed in Chapter 3 of the Skagit/Hanford PSAR.

In addition, further response to NRC staff questions satisfied the NRC that the staff position would be met, as documented in Subsection 3.10 of the SER for the Skagit Nuclear Power Project, dated September 1977 (NUREG-0309).

A-47 Safety Implications of Control Systems

Description

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 loss of a power supply, short circuit, open circuit, or sensor failure to cause simultaneous malfunction of several control features. Such an occurrence would conceivably result in a transient more severe than those transients analyzed as anticipated operational occurrences. A second concern is for a postulated accident to cause control system failures which would make the accident more severe than analyzed. Accidents could conceivably cause control system failures by creating a harsh environment in the area of the control equipment or by physically damaging the control equipment. Although it is generally believed that such control system failure would not lead to serious events or resulting conditions that safety systems cannot safely handle, in-depth studies have not been rigorously performed to verify this belief. The potental for an accident that would affect a particular control system, and effects of the control system failures, may differ from plant to plant. Therefore, it may not be possible to develop generic answers to these concerns, but rather plant-specific reviews may be required. The purpose of this "Unresolved Safety Issue" is to define generic criteria that will be used for plant-specific reviews. A specific subtask of this "Unresolved Safety Issue" will be to study the reactor overfill transient in boiling water reactors to determine the need for preventive and/or mitigating design measures to preclude or minimize the consequences of this transient. Additional subtasks may be developed and resolutions required.

Resolution

The Skagit/Hanford 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".

A-47 Safety Implications of Control Systems (con't)

This has been accomplished by either providing independence between safety and nonsafety systems or providing isolating devices between safety and nonsafety systems. These devices are designed to preclude the propagation of nonsafety system equipment faults such that operation of the safety system equipment is not impaired. Should additional subtasks be developed, their resolutions will be provided in the FSAR.

A systematic evaluation of the control system design, such as contemplated for this "Unresolved Safety Issue," has not been performed to determine whether postulated accidents could cause significant control system failures which would make the accident consequences more severe than presently analyzed. However, as described in Chapter 15 of the Skagit/Hanford PSAR, a wide range of bounding transients and accidents has been analyzed to assure that the postulated events would be adequately mitigated by the safety systems.

The subtask of this issue concerning the reactor overfill transient in boiling water reactors has been reviewed by the BWR Owner's Group. Notwithstanding ultimate resolution of this item Puget Power has incorporated in the Skagit/Hanford design (see Section 7.4.1.1.3.1 of the 251 NSSS GESSAR which is incorporated as part of the S/HNP PSAR) a commercial grade high level (level 8) trip to the RCIC, HPCS, and feedwater systems to prevent the occurrence of the overfill transient.

Additionally, as provided in Appendix 1B (item II.B.8(1)) of the S/HNP PSAR, Skagit/Hanford has committed to develop a risk assessment program to identify significant and practical improvements in the reliability of core and containment heat removal systems that do not impact excessively on the plant design. This program will include an investigation of common cause failure mechanisms such as environmental factors, operator or maintenance errors, passive failures and system interactions

Changes in the design of control systems can be accommodated prior to the issuance of the operating license since instrumentation design is normally completed in the latter stages of plant construction.

Additional subtasks will be addressed as they are issued.

A-48 Hydrogen Control Measures and Effects of Hydrogen Burns on Safety Equipment

Description

Following a loss-of-coolant accident in a light water reactor plant, combustible gases, principally hydrogen, may accumulate inside the primary reactor containment as a result of: (1) metal water reaction involving the fuel element cladding; (2) the radiolytic decomposition of the water in the reactor core and the containment sump; (3) the corrosion of certain construction materials by the spray solution; and (4) any synergistic chemical, thermal and radiolytic effects of post-accident environmental conditions on containment protective coating systems and electric cable insulation.

Because of the potential for significant hydrogen generation as the result of an accident, 10 CFR Section 50.44, "Standards for Combustible Gas Control System in Light Water Cooled Power Reactors," and Criterion 41 of the General Design Criteria, "Containment Atmosphere Cleanup," in Appendix A to 10 CFR Part 50, require that systems be provided to control hydrogen concentrations in the containment atmosphere following a postulated accident to ensure that containment integrity is maintained.

The regulation, 10 CFR Section 50.44, requires that the combustible gas control system provided be capable of handling the hydrogen generated as the result of degradation of the emergency core cooling system such that the hydrogen release is five times the amount calculated in demonstrated 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.

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.

Resolution

The S/HNP Containment Combustible Gas Control System designed to meet the requirements of 10 CFR 50.44 is

A-48 Hydrogen Control Measures and Effects of Hydrogen Burns on Safety Equipment (con't)

described in the S/HNP PSAR Section 6.2.5. This system was reviewed and found acceptable by the NRC Staff in the Skagit SER (NUREG-0309, September 1977) Section 6.2.16.

As detailed in the S/HNP PSAR Amendment 22 (Appendix 1B, item II.B.8.(3) and II.B.8(4), a hydrogen control system capable of handling hydrogen generated by the equivalent of a 100% active fuel-clad metal water reaction will be provided. The NRC Staff found in Supplement 2 to the S/HNP SER (NUREG-0309, Supplement 2, October 1981) that the commitments provided by the applicant with respect to the hydrogen control requirements are acceptable.