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DENTON,H.R. Office of Nuclear Reactor Regulation

SUBJECT: Responds to HR Denton 790917 ltr re safety grade sys.Has not identified interaction between nonsafety & safety grade sys constituting substantial safety hazard.Conclusion in FSAR that no undue risk to public safety exists is unchanged.

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

H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2
DOCKET NO. 50-261
LICENSE NO. DPR-23
HIGH-ENERGY LINE BREAK ENVIRONMENTAL EFFECTS ON CONTROL SYSTEMS

Dear Mr. Denton:

This letter responds to your September 17, 1979 letter on the subject of a "potential unreviewed safety question on interaction between non-safety grade systems and safety grade systems". This potential problem was further addressed in IE Information Notice 79-22, dated September 14, 1979.

In conjunction with Westinghouse, our NSSS Vendor , we have reviewed the specific non-safety grade systems listed in IE Information Notice 79-22, as well as others, for potential interactions that could constitute a substantial safety hazard. We have not been able to identify such an interaction. While, in some cases, we have identified variations from the FSAR licensing bases, the basic conclusion of the FSAR, that these events do not constitute an undue risk to the health and safety of the public, remains unchanged.

Scope

On September 18, 1979, Westinghouse presented to the Staff a summary of the investigation that had been conducted which led to the identification of four (4) generic potential interaction scenarios where the affect of adverse environments, resulting from high energy line breaks on control systems could possibly lead to consequences more limiting than the results presented in the Safety analysis Report. Table 1 summarizes the scope of the Westinghouse investigation.

The seven (7) control systems selected for the investigation by Westinghouse include all control systems directly addressed in the current Westinghouse functional requirements. The seven (7) accidents considered encompass all postulated High Energy Line Break (HELB) environments, including all break locations and a range of break sizes. Of the forty-nine (49) combinations of control system and accident environment investigated, fifteen (15) interaction scenarios, denoted by an X in Table 1, were identified which resulted in potential

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consequences more severe than reported in the Safety Analysis Reports. The fifteen interactions identified are bounded by consideration of the four (4) generic interactions which are discussed in attachments to this letter. Individual plant characteristics may make some of the postulated generic interactions not applicable to that plant. A specific discussion of the applicability of the four (4) bounding Westinghouse senarios to H. B. Robinson, Unit No. 2 is given below.

H. B. Robinson Specific

1. Steam Generator Power-Operated Relief Valve Control System

H. B. Robinson Unit No. 2 is an outdoor plant with the steam generator Power Operated Relief Valves (PORV's) located on the outside of the containment in an open environment. There is considerable physical separation between the PORV's and the main feedwater lines which are also located in an open area. Therefore, no adverse environment for either the steam generator PORV's or their control circuitry can be created by a rupture of the main feedwater line. The auxiliary feedwater lines run through the lower level of the Auxiliary Building, which is one level below the control circuit power supplies for all of the control equipment of concern. The levels are separated by a three-hour fire barrier, and, in any event, the auxiliary feedwater lines carry only cold water so no steam environment would be created in the event of a break. Additional conservatism exists in that the two motor-driven and one steam-driven auxiliary feedwater pumps are each of 100% capacity, and thus, any one pump will supply adequate cooling to the steam generators.

It does not appear that an adverse environment for the steam generator PORV's or their controls will exist as the result of an accident and, therefore, this concern does not appear to be applicable to H. B. Robinson Unit No. 2.

2. Main Feedwater Control System

As indicated above, the Robinson main feedwater lines do not run through the Auxiliary Building and thus cannot create an adverse environment there. The apparent concern in this event is that a malfunction of the feedwater control system could result in all steam generators being at low level before a reactor trip is obtained. This situation was analyzed in the H. B. Robinson Unit No. 2 FSAR and found to be acceptable.

It does not appear that the postulated adverse environment will exist for the main feedwater controls, and even if it could exist, the resultant situation has already been analyzed. This concern does not, therefore, appear to be applicable to H. B. Robinson Unit No. 2.

3. Pressurizer Power-Operated Relief Valve Control System

The H. B. Robinson pressurizer PORV's are spring-closed valves which require power and air to open. If an adverse environment or a signal should interrupt either the air or power supply, or if the solenoid valve diaphragm should fail, the PORV's will close. In addition, there is an existing interlock through the Safeguards System to deenergize the PORV solenoid valves which provide the air supply at a decreasing pressurizer pressure of 2000 psi.

All of the instrumentation on the H. B. Robinson pressurizer for control functions are of the same quality as the protective function instrumentation, and are, therefore, not expected to be adversely affected by the containment accident environment. The air supply solemoid valves are not environmentally qualified and will be replaced at the next refueling outage with qualified valves. In the interim, all operators will be alerted to the possibility of a consequential failure in the pressurizer PORV system and instructed to close the block valves in the relief lines if they believe a PORV has opened. The operator would be aware of a developing containment adverse environment from an alarm received on the condensate collecting system in the containment which would be received before the environment degraded to the point of damaging any equipment.

This concern is applicable to the H. B. Robinson Plant, but most of the equipment will not be affected by the accident environment. The solenoids will be replaced at the next refueling outage, and the increased operator awareness is a sufficient interim measure to safeguard the public health and safety.

4. Automatic Rod Control System

This item is concerned with a failure of the rod control system as the result of a steam line rupture inside containment which causes the rods to withdraw. Westinghouse has performed a generic evaluation of this sequence of events and has concluded that, even with initial rod withdrawal, there will be no fuel failure and thus no increase in risk to the public health and safety. In addition, all operators will be alerted to this potential consequential failure and directed to take appropriate corrective action such as rod insertion or reactor trip.

Probability of Postulated Interactions

Implicit in the four (4) generic potential interaction scenarios identified by Westinghouse are worst case assumptions concerning the break size and location, and the type and extent of consequential failures in control system induced by the adverse environment. These assumptions are therefore in addition to the already conservative set of assumptions ascribed to the analysis of the Design Basis

Events reported in the Safety Analysis Report. It follows that these scenarios represent a significantly less probable subset of the Design Basis Events that are dependent on the occurrence of additional events, each having an associated uncertainty of occurring. While no specific quantitative analysis has been conducted for H. B. Robinson Unit No. 2 concerning the improbability of overall scenarios, the attachments define, for each of the generic scenarios the conservative assumptions already contained in the Design Basis Event analysis reported in the Safety Analysis Report and the additional conservative assumptions to be made to derive the postulated interaction scenario.

As can be seen from the attachment, for each of the scenarios considered, the improbability of all the additional set of assumed conditions occurring simultaneously, over and above the already low probability of the Design Basis Event itself, leads to the conclusion that continued operation of H. B. Robinson Unit No. 2 can be justified until solutions to these low probability event scenarios can be implemented.

With regard to the probability of any single design basis event initiating, via the adverse environment, failures in several control systems, it again can be noted from the attachments that the probability of all the additional set of conditions occurring simultaneously for more than one scenario is of an even lower order of magnitude than for each individual scenario.

In addition to the qualitative probability discussion above, a specific quantitative probability analysis for a typical plant will be submitted to your staff by the Atomic Industrial Forum during the week beginning October 8, 1979. The analysis will be titled "Probalistic Analysis of IE Information Notice 79-22 Scenarios" and is being prepared by the Nuclear Safety Analysis Center of EPRI.

Future Activities

As you must recognize, our investigation within the limited time frame required by your September 17 letter must be considered preliminary and could not include detailed evaluations. Generic analyses coupled with plant-specific, detailed evaluations, where required, are proceeding, expeditously, and the results will be reported to you as discussed below. However, based on our preliminary investigation and results as discussed above, continued operation is warranted while these detailed evaluations proceed.

As a result of the Three Mile Island accident, there are a significant number of industry, governmental and regulatory investigations under way examining the licensing bases and the operating procedures of nuclear generating facilities. These investigations are already identifying areas where studies may result in the consideration of new or revised events as part of the bases for assuring the continuing safety of nuclear plants. NUREG-0578 outlines several such events and suggests remedies.

Accident \ Control System	Reactor Control	Pressure Control	Level Control	Feedwater Control	Steam Generator Pressure Control	Steam Dump System	Turbine Control
Small Steamline Rupture	X	X			X		
Large Steamline Rupture		X			X		
Small Feedline Rupture	X	X		X	X		
Large Feedline Rupture	X	X			X		
Small LOCA	X	X		X			
Large LOCA							
Rod Ejection							

TABLE 1

PROTECTION SYSTEM-CONTROL SYSTEM POTENTIAL ENVIRONMENTAL INTERACTION

X - Potential Interaction Identified that could Degrade Accident Analysis

\- No such Interaction Mechanism Identified

NUREG-0578 requirements for analyses of potential safety problems envision the kinds of scenarios identified by Westinghouse and made the subject of IE Information Notice 79-22.

We therefore, believe that the scope of the action required to resolve the concerns which were the subject of IE Information Notice 79-22 is consistent with the requirements of NUREG-0578 and should therefore be integrated with the planned response sequence for compliance with the NUREG.

If you have additional questions, please do not hesitate to call upon me or my staff.

Very truly yours,

M A McDuffie

for E. E. Utley
Executive Vice President
Power Supply & Customer Service

CSB:lgb*

Attachments

Sworn to and subscribed before me this 4th day of October, 1979.

My commission expires: October 4, 1981.

Franklin Murray
NOTARY PUBLIC

FRANKLIN MURRAY
NOTARY PUBLIC
WAKE CO[illegible] N.C.

ATTACHMENT I

STEAM GENERATOR PORV CONTROL SYSTEM

1. Summary of Postulated Scenario

Following a feedline rupture outside containment in the auxiliary building, the steam generator PORV's are assumed to exhibit a consequential failure due to an adverse environment. Failure of the PORV's in the open position results in the depressurization of multiple steam generators which are the source of steam supply for the turbine driven auxiliary feedwater pump. Eventually, the turbine driven auxiliary feedwater pump will not be capable of delivering auxilary feedwater to the intact steam generators. Depending upon auxiliary system design, a potential exists that no auxiliary feedwater will be injected into the intact steam generators until the operator takes corrective action to isolate the auxiliary flow spilling out the rupture.

2. Probability

Assumptions Affecting Event Probability and Consequences

a. Standard Safety Analysis Report Assumptions Concerning Feedline Break
   - conservative initial assumptions
     - o Appendix K decay heat model
     - o Engineered safeguards power plus calorimetric error
     - o programmed RCS temperature plus control deadband and instrument errors
     - o initial conservative S/G inventory
     - o conservative core physics

   - conservative accident assumptions
     - o Break (all sizes) om Safety Class 2 feedline piping
     - o maximum adverse environmental errors for protective instrumentation
     - o worst single active failure (loss of one motor-driven auxiliary feed pump)
     - o operator action time

b. Additional Assumptions Required for this Scenario

   - Break must occur outside containment between the penetration and feedline check valve.

   - Adverse environment resulting from the rupture can impact the steam generator PORV control systems associated with the ruptured loop and the intact loops.

- The single active failure is a motor driven auxiliary feed pump. The loss of a turbine driven auxiliary feed pump as the single active failure or no active failure would invalidate the postulated scenario.

- Due to the adverse environment, the steam generator PORV control system initiates a spurious signal to open the PORV(s). Should the control system continue to operate within specification or initiate a spurious signal to close the PORV(s), the scenario is invalidated.

- PORV on steam generators supplying steam to turbine driven auxiliary feed pump is assumed to open as a result of spurious signal. If this PORV is not affected or fails closed, the scenario is invalidated.

3. <u>Accident Consequences</u>

Section 4.2 of WCAP-9600, Report on Small Break Accidents for Westinghouse NSSS Systems, describes transient analyses for postulated loss of all main and auxiliary feedwater (no pipe rupture). The results indicate that the operator has at least 4,000 seconds following the loss of all feedwater to reinitiate auxiliary feedwater flow to the steam generators before the core begins uncovering.

The interaction scenario postulated above is similar to that presented in Section 4.2 of WCAP-9600. The only additional assumption made is that a feedline rupture occurs outside containment between the containment penetration and the feedline check valve. Conservatively assuming that all liquid inventory in the steam generator associated with the ruptured feedline is lost via the rupture without removing any heat (i.e., liquid blowdown), calculations have shown that the heat removal capability of the liquid inventory blowdown requires operator action 1200 seconds earlier than reported in WCAP-9600. Thus, if a feedline rupture is assumed coincident with the analyses performed in WCAP-9600, the operator still has at least 2800 seconds to take corrective action to inject auxiliary feedwater into the intact steam generators. No Safety Analysis Reports assume greater than 30 minute operator action following a feedline rupture.

ATTACHMENT II

MAIN FEEDWATER CONTROL SYSTEM

1. Summary of Postulated Scenario

Following a small feedline rupture, the main feedwater control system malfunctions in such a manner that the liquid mass in the intact steam generators is less than for the worst case presented in Safety Analysis Reports. The reduced secondary liquid mass at time of automatic reactor trip results in a more severe reactor coolant system heatup following reactor trip.

2. Probability

Assumptions Affecting Event Probability and Consequences

a. Standard Safety Analysis Report Assumptions Concerning Feedline Break

- conservative initial assumptions
    - o Appendix K decay model
    - o Engineered safeguards power plus calorimetric error
    - o Programmed RCS temperature plus control deadband and instrument error
    - o initial conservative S/G inventory
    - o conservative core physics

- conservative accident assumptions

    - o break (all sizes) in Safety Class 2 feedline piping
    - o maximum adverse environmental errors for protective instrumentation
    - o worst single active failure (loss of any one auxiliary feed pump)
    - o operator action time

b. Additional Assumptions Required for this Scenario

- Break must occur between S/G nozzle and feedline check valve. A break at any other location invalidates the scenario.

- Small breaks less than 0.2 sq. ft. Larger breaks invalidate the scenario.

- Adverse environment resulting from the break can impact both the main feedwater control systems associated with the broken loop and the intact loops.

- Due to the adverse environment, the main feedwater control system initiates a spurious signal to close the feedwater control valves (FCV) in the intact loops. Should the control system continue to operate within specification the scenario is invalidated.

3. <u>Accident Consequences</u>

Section 4.2 of WCAP-9600, Report on Small Break Accidents for Westinghouse NSSS System, describes transient analyses for a postulated loss of all main and auxiliary feedwater (no pipe rupture). Following a loss of all main and auxiliary feedwater, the operator is not required to take action for at least 4,000 seconds following the loss of all feedwater to prevent the core from uncovering. With a feedline rupture assumed coincident with the assumptions made in WCAP-9600, the operator continues to have at least 2800 seconds before corrective action must be taken to inject auxiliary feedwater into the intact steam generators to prevent core uncovering. No Safety Analysis Reports assume greater than 30 minute operator action follwing a feedline rupture.

ATTACHMENT III

PRESSURIZER PORV CONTROL SYSTEM

1. Summary of Postulated Scenario

   Following a feedline rupture inside containment, the pressurizer PORV control system malfunctions in such a manner that the power operated relief valves fall in the open position. Thus, in addition to a feedline rupture between the steam generator nozzle and the containment penetration, a breach of the reactor coolant system boundary has occurred in the pressurizer vapor space.

2. Probability

   Assumptions Affecting Event Probability and Consequences

   a. Standard Safety Analysis Report Assumptions Concerning Feedline Break

      - conservative initial assumptions
        - o Appendix K decay heat model
        - o Engineered safeguards power plus calorimetric error
        - o Programmed RCS temperature plus control deadband and instrument errors
        - o initial conservative S/G inventory
        - o conservative core physics

      - conservative accident assumptions
        - o break (all sizes) in Safety Class 2 feedline piping
        - o maximum adverse environmental errors for protective instrumentation
        - o worst single active failure (loss of any one auxiliary feed pump)
        - o operator action time

   b. Additional Assumptions Required for this Scenario

      - Break occur inside the containment between the steam generator nozzle and the containment penetration. A break at other locations invalidates this scenario.

      - Double ended break leads to limiting consequences. Smaller breaks permit longer operator action times.

        Adverse environment resulting from the break can impact the pressurizer power operated relief valve control system.

- Due to the adverse environment, the pressurizer PORV control system initiates a spurious signal to open the PORV(s). Should the control system continue to operate within specification or initiate a spurious signal to close the PORV's, the scenario is invalidated.

- Should the PORV's fail to the preset safe position (i.e. closed), the scenario is invalidated.

3. Accident Consequences

Section 4.2 of WCAP-9600, Report on Small Break Accidents for Westinghouse NSSS Systems, describes transient analyses for a postulated loss of all main and auxiliary feedwater (no pipe rupture). The results indicate that, in the event that the operator cannot restore auxiliary feedwater to the steam generators, the operator is required to open the presssurizer PORV's within 2,500 seconds to maintain adequate core coolant inventory.

The interaction scenario postulated above is similar to that presented in Section 4.2 of WCAP-9600. The additional assumptions made are the following:

a. a feedline rupture is assumed to occur between the steam generator nozzle and the containment penetration

b. auxiliary feedwater is injected into the intact steam generator following the feedline rupture.

Conservatively assuming that all liquid inventory in the steam generator associated with the ruptured feedline is lost via the rupture without removing any heat (i.e., liquid blowdown), the loss of heat sink due to the liquid inventory blowdown of the ruptured steam generator is more than counterbalanced by the auxiliary feedwater being injected into the intact steam generators following reactor trip. Therefore, the results of the analyses present in WCAP-9600, Section 4.2, which illustrates that the operator is not required to take corrective action for at least 2,500 seconds following the loss of feedwater also applies to this scenario. No Safety Analysis Reports assume greater than 30 minute operator action following a feedline rupture.

ATTACHMENT IV

ROD CONTROL SYSTEM

1. Summary of Postulated Scenario

Following an intermediate steamline rupture inside containment, the automatic rod control system exhibits a consequential failure due to an adverse environment which causes the control rods to begin stepping out prior to receipt of a reactor trip signal on overpower delta-T. This scenario results in a lower DNB ratio than presently presented in Safety Analysis reports.

2. Probability

Assumptions Affecting Event Probability and Consequences

a. Standard Safety Analysis Report Assumptions Concerning Steamline Break

- conservative initial assumptions
  - o nominal rated power plus calorimetric error
  - o Programmed RCS temperature plus control deadband and instrument errors
  - o conservative end of life core physics

- conservative accident assumptions

  - o break (all sizes) in Safety Class 2 steamline piping
  - o maximum adverse environmental errors for protective instrumentation
  - o worst single active failure (loss of any one Safety Injection pump)
  - o operator action time

b. Additional Assumptions Required for this Scenario

- Break must occur inside the containment between the steam generator nozzle and the containment penetration. A break at other locations invalidates this scenario.

- Intermediate steamline breaks (0.1 to 0.2 sq. ft. per loop) at power levels from 70 to 100 percent. Other break sizes and power levels invalidate the scenario.

- Adverse environment from the break can impact the nuclear instrumentation system (NIS) equipment (i.e. excore neutron detectors, cabling connectors, etc.) prior to reactor trip (i.e. within 2 minutes). Should the NIS equipment not be affected until after reactor trip (i.e. later than 2 minutes), the scenario is invalidated.

- Due to the adverse environment, the NIS system initiates a spurious low power signal without causing a reactor trip on negative flux rate. Should the NIS continue to operate within specification, initiate a spurious high power signal, or cause a reactor trip on negative power rate, the scenario is invalidated.

3. <u>Accident Consequences</u>

A typical bounding analysis of the intermediate steamline rupture was performed to calculate the extent of fuel damage due to rod control system withdrawal prior to reactor trip. Based upon the reduction in radial peaking factor with burn-up and conservative end-of-life physics parameters, no fuel damage was calculated to occur following the intermediate steamline rupture with a consequential rod control system failure.