REGULATORY INFORMATION DISTRIBUTION SYSTEM (RIDS)

ACCESSION NBR:8201260194 DOC.DATE: 82/01/13 NOTARIZED: NO DOCKET #
FACIL:50-397 WPPSS Nuclear Project, Unit 2, Washington Public Powe 05000397
AUTH.NAME AUTHOR AFFILIATION
BOUCHEY,G.D. Washington Public Power Supply System
RECIP.NAME RECIPIENT AFFILIATION
SCHWENCER,A. Licensing Branch 2

SUBJECT: Forwards 120-day response to NRC 810424 concerns re pipe breaks in BWR scram discharge vol & response to Generic Ltr 81-34 re breaks.

DISTRIBUTION CODE: B001S COPIES RECEIVED:LTR 1 ENCL 1 SIZE: 10
TITLE: PSAR/FSAR AMDTS and Related Correspondence

NOTES:2 copies all matl:PM. 05000397

| | RECIPIENT ID CODE/NAME | | COPIES LTTR | COPIES ENCL | RECIPIENT ID CODE/NAME | | COPIES LTTR | COPIES ENCL |
|---|---|---|---|---|---|---|---|---|
| ACTION: | A/D LICENSNG | | 1 | 0 | LIC BR #2 BC | | 1 | 0 |
| | LIC BR #2 LA | | 1 | 0 | AULUCK,R. | 01 | 1 | 1 |
| INTERNAL: | ELD | | 1 | 0 | IE | 06 | 3 | 3 |
| | IE/DEP/EPDB | 35 | 1 | 1 | IE/DEP/EPLB | 36 | 3 | 3 |
| | MPA | | 1 | 0 | NRR/DE/CEB | 11 | 1 | 1 |
| | NRR/DE/EQB | 13 | 3 | 3 | NRR/DE/GB | 28 | 2 | 2 |
| | NRR/DE/HGEB | 30 | 2 | 2 | NRR/DE/MEB | 18 | 1 | 1 |
| | NRR/DE/MTEB | 17 | 1 | 1 | NRR/DE/QAB | 21 | 1 | 1 |
| | NRR/DE/SAB | 24 | 1 | 1 | NRR/DE/SEB | 25 | 1 | 1 |
| | NRR/DHFS/HFEB | 40 | 1 | 1 | NRR/DHFS/LQB | 32 | 1 | 1 |
| | NRR/DHFS/OLB | 34 | 1 | 1 | NRR/DHFS/PTRB | 20 | 1 | 1 |
| | NRR/DSI/AEB | 26 | 1 | 1 | NRR/DSI/ASB | 27 | 1 | 1 |
| | NRR/DSI/CPB | 10 | 1 | 1 | NRR/DSI/CSB | 09 | 1 | 1 |
| | NRR/DSI/ETSB | 12 | 1 | 1 | NRR/DSI/ICSB | 16 | 1 | 1 |
| | NRR/DSI/PSB | 19 | 1 | 1 | NRR/DSI/RAB | 22 | 1 | 1 |
| | ~~NRR/DSI/RSB~~ | 23 | 1 | 1 | NRR/DST/LGB | 33 | 1 | 1 |
| | REG FILE | 04 | 1 | 1 | | | | |
| EXTERNAL: | ACRS | 41 | 16 | 16 | BNL(AMDTS ONLY) | | 1 | 1 |
| | FEMA-REP DIV | 39 | 1 | 1 | LPDR | 03 | 1 | 1 |
| | NRC PDR | 02 | 1 | 1 | NSIC | 05 | 1 | 1 |
| | NTIS | | 1 | 1 | | | | |

TOTAL NUMBER OF COPIES REQUIRED: LTTR ~~63~~ 65 ENCL ~~58~~ 60

**Washington Public Power Supply System**
P.O. Box 968 3000 George Washington Way Richland, Washington 99352 (509) 372-5000

January 13, 1982
G02-82-37
SS-L-02-CDT-82-017



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
RESPONSES TO REQUEST FOR INFORMATION

Enclosed are sixty (60) copies of the Supply System 120-day response to the NRC's concern regarding Pipe Breaks in the BWR Scram Discharge Volume (see Reference 1 to the Attachment).

Also, this is the Supply System's response to Generic Letter 81-34 (see Reference 4 to the Attachment).

Very truly yours,

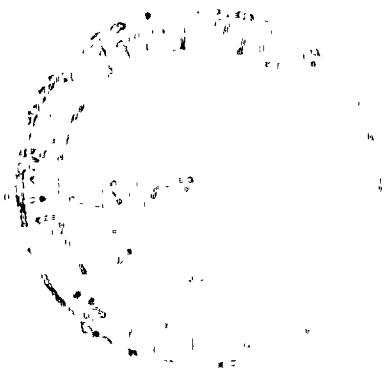
G D Bouchey

G. D. Bouchey
Deputy Director, Safety and Security

CDT/jca
Enclosures

cc: R Auluck - NRC
WS Chin - BPA
R Feil - NRC Site

B001
6/1



ATTACHMENT

Safety Concerns Associated with Pipe Breaks

in the BWR Scram System

References:

1. Letter from R. L. Tedesco (NRC) to R. L. Ferguson (Supply System), "Safety Concerns Associated with Pipe Breaks in the BWR Scram System", dated April 24, 1981.

2. GE Evaluation in Response to NRC Request Regarding BWR Scram System Pipe Breaks, NEDO-24342, dated April, 1981.

3. Letter MFN-091-81, G. C. Sherwood (GE) to D. Eisenhut (NRC), "NRC Report, 'Safety Concerns Associated with Pipe Breaks in the BWR Scram System'", dated April 30, 1981.

4. Generic Letter 81-34, Darrell G. Eisenhut (NRC) to All GE BWR Licensees, same title, dated August 31, 1981, NUREG 0803, enclosed.

## 1.0 INTRODUCTION

In response to your request for a generic evaluation within 45 days of the Scram Discharge Volume System, (SDV) (Reference 1), GE reviewed the SDV against the general design criteria and it was concluded that the generic SDV design is in conformance with GDC 14, GDC 35, GDC 55, 50.2(v), 50.55a (including footnote 2), and § 50.46 of the Commission's regulations. The generic evaluation report (Reference 2) was transmitted to you on April 30, 1981 (Reference 3).

In addition to your request for the generic evaluation, you also requested, within 120 days, a plant specific evaluation of the applicability of the 45 day generic evaluation. The WNP-2 SDV has been reviewed on a plant specific basis by a team of Supply System engineers using AE input and confirmed by a plant walkdown. The generic report was found to envelop the WNP-2 plant design. Further NRC guidance was provided in NUREG 0803 (Reference 4) as to an acceptable plant specific, 120 day response for this issue. The Supply System response is enclosed in a format compatible with Section 5 of NUREG 0803, i.e., after a section describing recent or proposed SDV changes, the following sections describe our response to Piping Integrity (Section 3.0), Mitigation Capability (Section 4.0), and Equipment Qualification (Section 5.0).

## 2.0 SYSTEM DESIGN

The WNP-2 SDV has been evaluated against the Generic Safety Evaluation Report, "BWR Scram Discharge System", dated December 1, 1980. The evaluation indicated that the installed system design satisfies the intent of the GE Generic Safety Evaluation Report and GE Design Specification 22A4260. The following changes will be required for compliance.

1. The addition of redundant air-operated vent and drain isolation valves.

2. The addition of six additional redundant and diverse level instrumentation for scram.

3. The relocation and repiping of instrument piping directly to the scram instrument volume.

These changes are discussed in our detailed response to NRC Question 010.41. (Attached)

WNP-2's SDV header system is designed as a continually expanding path from the 185 3/4" individual scram discharge (withdrawal) lines to one of two integrated SDV/IV (Instrument Volume) systems (one system per approximately half the drives). Each integrated SDV/IV system consists of a continuously downsloping piping run expanding from the SDV (consisting of seven 6" return headers from the individual hydraulic control unit (HCU) banks to an 8" combined return header) to the 12" vertically oriented IV. WNP-2's IVs have been designed as vertical extensions attached directly to the SDV. This configuration provides a direct hydraulic couple between the SDV and IVs and ensures immediate and continuous liquid level monitor in the SDV.

Redundant air-operated vent and drain valves will be added on the SDV in series to ensure system isolation during reactor scrams. This includes independent solenoid valves for each set of air-operated vent and drain valves.

The SDV is designed with an integral IV which provides direct and immediate detection of liquid accumulation. The SDV instrumentation is redundant and single-failure proof (including partial loss of service functions). Each IV is provided with high level and rod block instrumentation attached directly to it to alert the operator to liquid accumulation in the SDV.

The system logic is designed in a one out of two, twice configuration. Each of the associated instrument channels is capable of being separately isolated for maintenance, testing or calibration without inadvertently scramming the reactor.

All the WNP-2 SDV instrumentation will be relocated and repiped directly to the IV instead of the vent and drain piping. Procedures will be modified to include functional testing of SDV level instrumentation after each scram. Six additional diverse level sensors will be added to the SDV system to ensure diversity and redundancy in level monitoring and scram functions. The addition of the level switches described above and periodic surveillance testing of the instruments will provide a continuous means of monitoring the SDV liquid level and ensuring instrument reliability.

## 3.0 PIPING INTEGRITY

NUREG 0803 recommendations in this area include as-built inspection, seismic analysis, procedure review, and in-service inspection. Each topic is discussed separately below.

3.1 Recommendation/Guidance - Provide the results of or a schedule for as-built inspection of the SDV/HCU piping and supports.

WNP-2 Response:

The current schedule to recieve as-built inspection results for the SDV/HCU piping and supports is second quarter of 1983. These results will be included as part of the turn-over package for the CRD system in the form of verified (QA/QC'd) drawings. See Supply System response to IE Bulletin 79-14 for further discussion of the as-built procedure.

3.2 Recommendation/Guidance - Provide any seismic reanalysis of the SDV piping and supports conducted in accordance with IE Bulletins (specifically 79-14) or otherwise.

WNP-2 Response:

The WNP-2 SDV/HCU system has been designed, built and inspected to the requirements of Seismic Category I. Therefore, as detailed in our previous response to IE Bulletin 79-14 and in our response to NRC Question 010.41, no further seismic re-analysis is necessary.

3.3 Recommendation/Guidance - Provide a schedule and program for reviewing and revising, as appropriate, the HCU-SDV maintenance, surveillance, and modification procedures.

WNP-2 Response:

The surveillance and maintenance procedures are currently being written and approved for WNP-2. The WNP-2 operational staff has been notified of this concern and will assure adequate precautions and sufficient guidelines are contained to minimize the potential for loss of the SDV system integrity when such integrity should be available.

3.4 Recommendation/Guidance - Provide a program of periodic inservice inspection for the SDV system meeting the requirements for Class 2 piping in the Section XI ASME Code.

WNP-2 Response:

The inservice inspection for the CRD Scram Discharge Headers will consist of a visual examination for evidence of leakage which will be performed in accordance with ASME Section XI.

All piping connecting to the scram discharge headers is less than 4" nominal pipe size and subject only to a visual examination for evidence of leakage. Furthermore, the headers are on the discharge side of the scram discharge system and their failure would not prevent a scram. The only connections with the RPV are via the 3/4" CRD tubing. A minimal amount of leakage due to water bypassing the drives during scram will occur, but following scram the leakage will essentially cease upon closure of the scram valves. In any event, the sum of the bypass leakages from all drives is anticipated to be much less than the reactor makeup system capacity. Therefore, a failure of a header or any connecting piping would not impair the ability to makeup the resulting loss of reactor water using normal makeup systems. A visual examination for evidence of leakage is, therefore, adequate and commensurate with the low operability and safety implications of a loss of integrity of that pipe.

## 4.0 MITIGATION CAPABILITY

NUREG 0803 recommendations in this area include addressing the BWR emergency operating procedures and permissible coolant activity. Each topic is discussed separately below.

4.1 Recommendation/Guidance - Commit to revising the BWR Owner's Group Emergency Procedure Guidelines to instruct the operator into a controlled blowdown of >100°F/hr when a trip occurs with no reset and coincident with indication of a leak in the reactor building. The rate of depressurization must be compatible with the qualification of equipment in the reactor building.

WNP-2 Response:

A break in the scram discharge volume does not present a new event requiring special training and is handled in accordance with existing procedures for the small break LOCA event. Also, since WNP-2 will have symptom-based emergency procedures, a specific problem such as this will not be addressed in the procedures. Should such a problem arise, however, it will be covered under existing procedures by virtue of its effect on plant systems.

The compatibility of the equipment qualification with the time frame implied by this cooldown rate is discussed in Section 5.0 of this response, as is the flooding concern.

4.2 Recommendation/Guidance - Implement the Standard Technical Specification for coolant activity.

WNP-2 Response:

The pipe break in the SDV piping system is a low probability event with low offsite consequences. Hence, the Supply System does not agree with the recommendation in NUREG 0803. Our technical specifications will be submitted prior to fuel load and contain a coolant activity limit consistent with the WNP-2 safety and environmental evaluations.

Immediate operator entry into the SDV/HCU area is not required. The preferred method for mitigating a postulated leak or rupture would be a) scram reset, or b) depressurization such that the leak rate, and hence, dose rate, is small. This would allow time for later personnel entry to manually isolate the leak or rupture.

## 5.0 ENVIRONMENTAL QUALIFICATION

NUREG 0803 recommendations include identification of equipment used to detect and to mitigate a break and/or leak in the SDV system. Each topic is discussed separately below.

5.1 Recommendation/Guidance - Identify the equipment that would be used to detect a break and/or leak in the SDV system and include the qualification of this equipment in the NRC's ongoing EQ program.

WNP-2 Response:

Using a symptom-based approach to the emergency procedures means that identification of a SDV leak or rupture is not the operator's primary concern. However, the Supply System agrees that early detection and mitigation is desirable to limit the amount of coolant spilled.

It is important to note the reactor conditions in that this is the atmosphere in which the operator makes his diagnosis (i.e., the reactor is tripped but unable to reset the scram, and greater than normal makeup is required to maintain vessel level). The primary indication will be from the Reactor Building CRD Hydraulic Equipment Area Radiation Monitors (Area "E"--Station 4, ARM-RE-4 and Area "W"--Station 5, ARM-RE-5). These monitors operate at an estimated background reading of 1 mR/hr and would increase in reading by approximately .17 mR/hr per gallon of coolant spilled. This would be an immediate and localized response to a SDV/HCU leak or rupture.

In addition, the equipment drain system at WNP-2 contains leak flow detectors at each floor level of the Reactor Building (Ref: FSAR, Fig. 9.3-5). The signal from these detectors are displayed and annunciated in the control room. This provides the operator with unambiguous indications of which floor level the leak started on.

As discussed in NUREG 0803, further diagnostic indicators would be CRD temperature and CRD position indications. Also, following a normal scram, certain SDV/IV temperature, pressure, and level indications would be expected. These indications would not necessarily be present or would be off-normal with a SDV pipe break, again pinpointing the SDV/HCU as the trouble-spot.

The large number of diverse, redundant signals available for diagnosis of a SDV leak or pipe break render environmental qualification of any single indicator as non-productive in increasing plant safety.

5.2 Recommendation/Guidance--Identify the equipment needed to mitigate an unisolable break in the SDV system and include its qualification in the NRC's ongoing EQ program; address specifically:

1) Equipment used for depressurization,
2) ECCS equipment designed for water impingement,
3) ECCS equipment qualified for wetdown by 212oF water,
4) feedwater and condensate system operation independent of the reactor building environment,
5) availability of HPCI-LPCI turbines due to high ambient temperature trips,
6) essential components qualified for service at 212oF and 100% humidity.

WNP-2 Response:

The preferred mitigation of a pipe break in the SDV system is for the operator to achieve a scram reset, thereby isolating the flow path from the break opening. There are several means of bypassing or temporarily disabling scram signals in the control room to achieve scram reset and hence, isolation.

If scram reset cannot be achieved, then reactor vessel depressurization is used to greatly reduce the break mass flow rate until the operators can manually close the isolation valves on the HCU. In order to depressurize and continue to remove decay heat, the following equipment is needed or could be used:

1) Equipment used for Depressurization--The equipment essential for depressurization is a means for steam relief and a source of make-up water. For steam relief, the operator can use either the turbine bypass system or, if that's not available, the safety-relief valves. Both of these systems are independent of a postulated SDV pipe break environment.

For make-up flow, the operator can use the feedwater system or the ECCS (HPCS and LPCS) systems. Their environmental qualification for a SDV pipe break are discussed in 2), 3), 4), and 5) below.

2) ECCS Equipment Designed for Water Impingement--The pump motors for the ECCS pumps are a drip-proof design, i.e., they are shrouded for protection against water impingement.

3) ECCS Equipment Qualified for Wetdown by 212oF Water--The Supply System has made a plant specific calculation using the methods found in Reference 2. The resulting flow rates and flooding data are tabulated below. Our analysis is in agreement with the conclusions and extends beyond the results presented in Reference 2. We agree with Reference 2 that 30 minutes is more than adequate time for the operator to recognize the situation and begin manual depressurization. Since the SDV piping collects the seal leakage flow from all CRDs the worst case break would occur at Floor El. 522', 100'3" above the reactor building basement where ECCS pumps are located, releasing initially 550 gpm from reactor vessel inventory. Since analysis of this break size spectrum shows that the release is ~30% steam (Reference 2), the following results are conservative in assuming that the flow is 100% sub-cooled water flowing directly to the ECCS pump rooms for the flooding condition.

Postulated SDV Pipe Break, Depressurization at

| | | | 6 Sumps Operating | | No Sumps Operating | |
|---|---|---|---|---|---|---|
| Time After Break (Min) | RPV Pressure (psia) | Rupture*** Flow (gpm) | Water Holdup** on Floor (gal) | Depth on Floor (in) | Water Holdup on Floor (gal) | Depth on Floor (in) |
| 0 | 1050 | 550 | 0 | 0 | 0 | 0 |
| 30 | 1050 | 550 | 7,500 | 0.24 | 16,500 | 1.5 |
| 40 | 135 | 130 | 7,900 | 0.30 | 19,900 | 2.0 |
| 70 | 89 | 73 | 1,945 | 0 | 22,945 | 2.4 |
| 180* | 23.8 | 45 | 0 | 0 | 29,435 | 3.3 |

* Leak isolated at 3 hrs. (2-1/2 hours after depressurization per Reference 2)

** The floor noted covers RHR pump rooms A and B, the CRD pump room and the radwaste room; the ECCS drive motors, the only items possibly susceptible to water damage are more than 60" above the floor.

*** The flow rate curve was integrated to give "water holdup" and "depth on floor".

The Reference 2 analysis notes that coolant activity and steam environment surrounding the HCU's following the break shows that 2-1/2 hours after depressurization begins is more than adequate time to allow operator access to close the HCU isolation valves.

The above results assume all watertight doors which separate ECCS compartments in the basement remain closed. However, in the above analysis, if any of these doors are left open, the effect would be to reduce the water depth on the floor and extend the time available for manual isolation of the HCU's. This is a consequence of simply increasing the floor area available for flooding. The only ECCS equipment susceptible to flooding is approximately 60" above the basement floor. HCU isolation could occur as late as 20 hours and still not flood ECCS pump motors even in the event no sump pumps were running.

In WNP-2 the electrical, instrument and control equipment which is vital to the ECCS and which is located in the reactor building is designed to seal against penetration by steam.

The foregoing establishes the fact that the ECCS pump capability will not be lost due to flooding, and this coupled with the availability of other vessel injection loops meets the requirements of 50.46 of the Commission's regulations.

4) Feedwater and Condensate System Operation Independent of the Reactor Building Environment - The only active components of the feedwater and condensate system operation that are exposed to the reactor building environment are the isolation valves RFW-V-065 A and B and the check valves RFW-V-32 A and B. Since these valves and associated signals are part of the Reactor Building Isolation System, they are Quality Class I, Seismic Category I components. The motor operators for these valves are part of the Supply System Environmental Qualification Program.

5) Availability of HPCI-LPCI Turbines due to High Ambient Temperature Trips - WNP-2 utilizes HPCS-LPCS systems with motor-driven, not turbine-driven, pumps. These motors do not receive a trip signal from an area high temperature sensor and, therefore, are available in an adverse environment.

6) Essential Components Qualified - The sections of NUREG 0803 applicable to this item discuss the availability of long-term cooling following an SDV pipe break. At WNP-2 the RHR system

is Quality Class I, Seismic Category I, Class 1E. Therefore, components of this system are a part of the Supply System Environmental Qualification Program.

The above discussion demonstrates that adequate equipment exists, under SDV pipe break conditions, to mitigate the break flow and continue core cooling.

5.3 Recommendation/Guidance - For any equipment required for identification and/or mitigation that is not qualified for service at 212°F and 100% humidity, provide a schedule for defining the plant-specific SDV break environment and a commitment to qualify the equipment in accordance with the NRC's ongoing EQ program.

WNP-2 Response:

The responses provided in the previous sections have demonstrated that mitigation and long-term cooling can be achieved with qualified equipment following a pipe break in the SDV/HCU system. Therefore, no further commitment of equipment to the EQ program is necessary. The detection systems employed, although not environmentally qualified, are numerous and redundant, alleviating the necessity of qualification for a low probability event.