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SCHWENCER,A. Licensing Branch 2

SUBJECT: Forwards marked-up,response to Containment Sys Branch Issue 47a-f,"Acceptable SRV Design Load Spec" from 810914-17 meeting.Revised Page 5 to response to Issue 48,submitted 820113,encl.

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January 28, 1982
G02-82-120
SS-L-02-CDT-82-024

Docket No. 50-397

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

Dear Mr. Schwencer:

Subject: NUCLEAR PROJECT NO. 2
RESPONSE TO CSB ISSUES 47 & 48

Enclosed are sixty (60) copies of the WNP-2 response to CSB Issue 47a-f, "Acceptable SRV Design Load Specification", from the NRC meeting held September 14-17, 1981. The attached response was inadvertently omitted from letter G02-82-35, dated January 13, 1982.

Also, enclosed are sixty (60) copies of revised page 5 to the WNP-2 response to CSB Issue 48, submitted to the NRC via G02-82-35, dated January 13, 1982.

Very truly yours,

for

G. D. Bouchey
Deputy Director, Safety and Security

CDT/jca
Enclosures

cc: R Auluck - NRC
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CONTAINMENT SYSTEMS BRANCH

ISSUE 47 Acceptable SRV Design Load Specification

NRC: Acceptance of pressure from Caorso based on 90/90 (9.37 psi) for subsequent actuation with the resolution of the following:

a. Use DFFR correlation to determine a Δ pressure rather than multiplier to account for differences between Caorso test conditions and WNP-2 design conditions, using the WNP-2 design temperature as specified in the Technical Specifications to calculate the Δ pressure.

b. Either account for differences in discharge line air volume between Caorso and WNP-2 or provide justification.

c. Account for the differences in pressure between the multiple and single valve case by adding the difference between the mean pressure from Caorso multiple valve case and the single valve case to the 9.37 psi, or provide justification that the WNP-2 design is conservative.

d. Account for effect of two vacuum breakers vs. one vacuum breaker as used in Caorso by adding the difference between the mean pressure from Caorso test conducted with two vacuum breaker and the tests conducted with one vacuum breaker blocked to the 9.37 psi or provide justification.

e. 1) For multiple valve case at initial actuation, the vertical pressure distribution used as specified in the WNP-2 SRV report is acceptable.

2) For single valve case, at subsequent actuation, the staff requires the method of NUREG-0487 for vertical pressure distribution or provide justification that the present load design is conservative.

f. Circumferential pressure distribution should be based on NUREG-0487 or as calculated from the Burns & Roe hydrodynamic model.

g. The staff requires in-plant tests (NUREG-0763) to verify ΔT between the bulk and local pool temperature and to verify boundary pressure loads.

Supply System: The Supply System will provide either resolution or justification by December 15, 1981.

Response to Issue 47 from Containment Systems Branch meeting with Washington Public Power Supply System at Richland, Washington on September 14-17, 1981.

ISSUE 47 Acceptable SRV Design Load Specification

Response

During the meeting of September 14-17, 1981 in Richland, Washington, between the Supply System and the Nuclear Regulatory Commission - Containment Systems Branch, it was agreed that the SRV design load specification (Issue 47) is acceptable provided six NRC detailed comments regarding subsequent actuation design load specification were either adopted or justification supplied to prove that the Supply System's present position is adequate. The six comments are addressed below in the same order as in the meeting minutes and the Supply System position for each is defined.

a. The Supply System agrees to use a pressure differential (ΔP) calculated with the DFFR Rev. 3 correlation in order to account for differences between CAORSO plant test conditions and WNP-2 design conditions. ~~With the suppression pool temperature at 120°F, as defined in the Technical Specifications for WNP-2, the calculated pressure differential equals 2.91 psi.~~

b. The safety relief valve discharge line (SRVDL) 1B air volume of 57.2 ft^3 (see Response to NRC Question 022.109, Table Q 022.109) is smaller than CAORSO's quencher A SRVDL air volume of 66 ft^3 (Reference 1.) by approximately 13%. Results of a second order regression analysis, available to the NRC, appear to indicate that larger "bubble" pressures (than obtained from the DFFR linear regression formula) may correspond to such smaller air volume values. Quencher 1-B is one of the two lowest pressure set point quenchers and is located adjacent to the RPV pedestal in WNP-2. Any increase in "bubble" pressure will be more than offset by the pressure attenuation with distance (calculated at 55.2%) and, as a result, will have no impact on the containment vessel structure. There are three other SRVDLs, 1A, 1C and 1D, with air volumes smaller than CAORSO's quencher A SRVDL by approximately 1.4%, 5.5% and 4.7%, respectively. These SRVDLs correspond to lower pressure set point quenchers located adjacent to the RPV pedestal as well and, consequently, will have no impact on the containment vessel structure. The remaining fourteen SRVDLs have air volumes larger than the CAORSO quencher A SRVDL. In view of the above, the Supply System concludes that there is no need to account for differences in discharge line air volumes between CAORSO's quencher A SRVDL and WNP-2 SRVDLs 1B, 1A, 1C and 1D.

Assuming a Technical Specification limit on suppression pool bulk temperature of 110°F ~~for continued operation at low power~~, the calculated pressure differential equals 2.31 psi. (This number is tentative pending establishment of the actual pool temperature limit in the Technical Specifications for WNP-2.)

c. The design basis operational SRV actuation events investigated for WNP-2 and listed in Table I attached indicate that only six lower pressure set-point valves may discharge at subsequent actuation conditions. These valves correspond to quenchers which are all located adjacent to the RPV pedestal and, in view of the pressure attenuation with distance (in excess of 50%), will have no impact on the containment vessel structure. The Supply System concludes that, in view of the above, there is no need to account for differences in pressure between the multiple and single valve discharge events at subsequent actuation conditions.

d. Please refer to the Supply System answers to NRC questions 022.054 and 022.057. As discussed in those answers, the characteristics and the number of vacuum breakers installed influence the reflood transient in the SRVDL which in turn establishes the initial water level in the line, i.e., an initial condition for the SRV discharge event. Test data for a diversity of initial water level conditions (low, normal and high) were gathered during the CAORSO tests and the load definition derived envelopes (statistically) the data. The CAORSO test matrix includes only four subsequent actuations with both vacuum breakers operating (Tests 22A02 through 22A05, with quencher "U") a number too small to provide a basis for statistical conclusions. However, it is of interest to note that the maximum pressure amplitude recorded during these four tests (9.4 psi) is about equal to the statistically derived design value (9.37 psi) and furthermore that the latter envelopes all CAORSO data (initial and subsequent).

Notwithstanding the above, the Supply System agrees as a measure of added conservatism to increase the statistically derived design value of +9.37 psi by +1.84 psi, which represents the difference between the mean value for subsequent actuation tests with two vacuum breakers functioning (+7.28 psi) and the mean value for subsequent actuation tests with only one vacuum breaker functioning (+5.44 psi).

e. 1) No comment (present definition acceptable).

2) Please refer to the Supply System answer to NRC question 022.059. As discussed there (refer to Figs. Q.022.059-1 through Q.022.059-4, ~~copies attached)~~ the vertical pressure distribution presented in the SRV report (Reference 2., Fig. 3.8a) is conservatively representative of the CAORSO test data. This fact was also verified by data from Tokai-2 in plant SRV actuation

tests (Reference 2, Fig. 3.8b, ~~copy attached)~~ as well as by the KARLSTEIN SRV actuation test data (refer to Figure 8.167 Rev. 1 dated 3/79 of the Design Assessment Report for Pennsylvania Power & Light Company's Susquehanna Steam Electric Station, Units 1 and 2).

Notwithstanding all the above, the Supply System agrees as a measure of added conservatism to adopt the excessively conservative vertical pressure distribution recommended by NRC in a manner compatible with presently implemented analyses, as illustrated in attached Fig. I. This ensures that the total pressure-load applied to the suppression pool vertical boundaries is preserved and requires an increase by 10.7% in boundary pressure.

f. No comment (present definition acceptable).

References

1. General Electric Company Report, "CAORSO SRV Discharge Tests Phase I Test Report," NEDE-25100-P, May 1979.

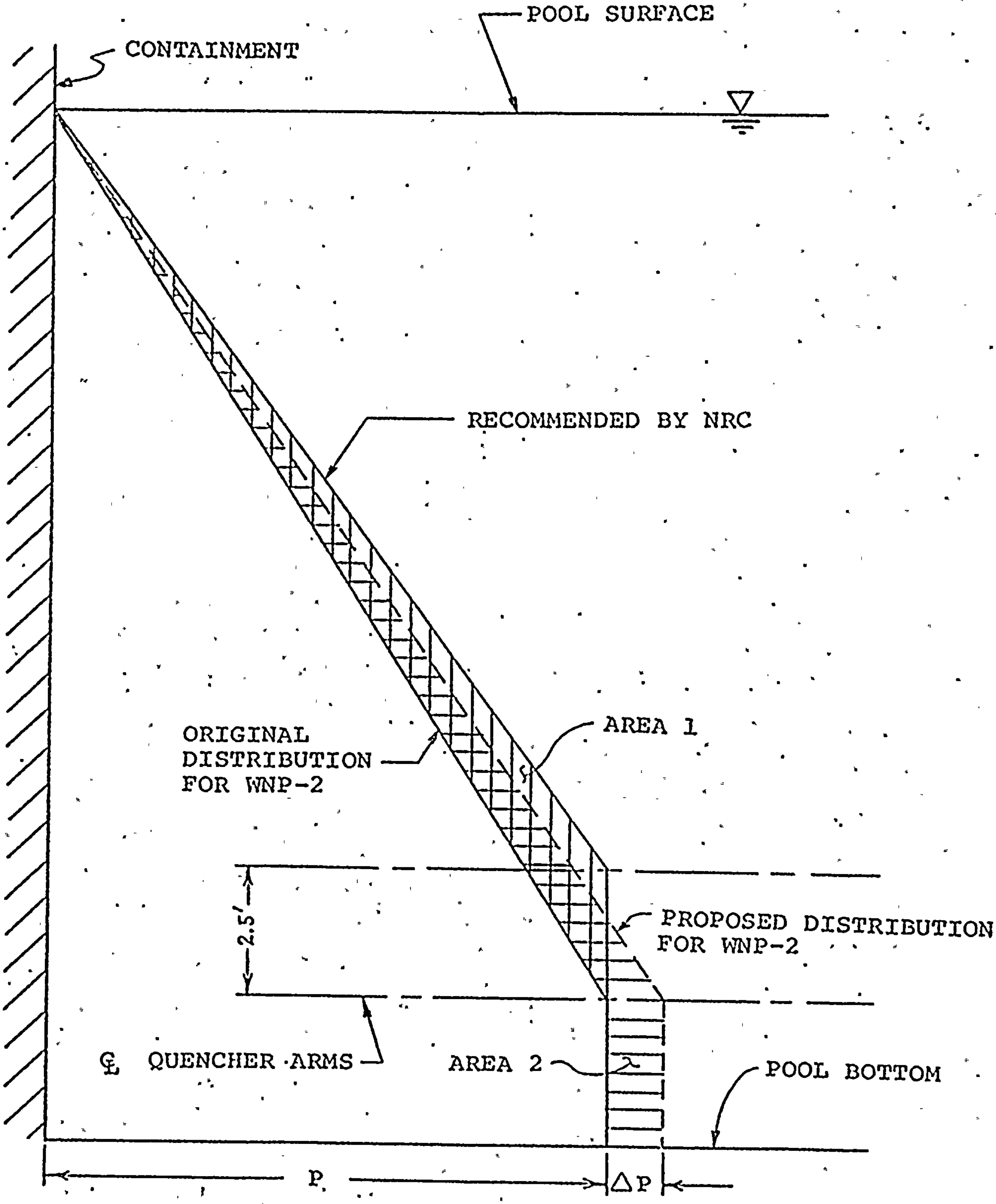
2. Burns and Roe, Inc., "SRV Loads - Improved Definition and Application Methodology for Mark II Containments."

3. Japan Atomic Power Company, "Tokai-2, Main Steam Safety Relief Valve Operational Test Report," August 1978.

TABLE I

WNP-2 DESIGN BASIS OPERATIONAL SRV ACTUATIONS

ITEM	FSAR SECTION	EVENT	NO. OF SRVs ACTUATED	
			1st BLOWDOWN	2nd BLOWDOWN
1	15.1.2	Feedwater Controller Failure	18	2
2	15.1.3	Pressure Regulator Failure-Open	2	2
3	15.1.4	Inadvertent SRV Opening	1	-
4	15.2.1	Pressure Regulator Failure-Closed	18	2
5	15.2.2	Generator Load Rejection-Bypass On	18	2
6	15.2.2	Generator Load Rejection-Bypass Off	18	2
7	15.2.3	Turbine Trip-Bypass On	18	2
8	15.2.3	Turbine Trip-Bypass Off	18	2
9	15.2.4	MSIV Closures	18	6
10	15.2.5	Loss of Condenser Vacuum	18	6
11	15.2.6	Loss of Auxiliary Power Transformers	2	2
12	15.2.6	Loss of All Grid Connections	18	2
13	15.2.7	Loss of Feedwater Flow	2	2
14	15.3.1	Trip of Both Recirculation Pumps (One Main Valve)	6	2
15	15.3.2	Recirculation Flow Control Failure (Both Main Valves)	2	2
16	15.3.2	Recirculation Flow Control Failure	6	2



FOR AREA 1 = AREA 2, $\Delta P = 10.7\% (P)$

WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	VERTICAL DISTRIBUTION OF POOL BOUNDARY PRESSURE	FIGURE I

LOCA STEAM CONDENSATION LOADS ON SUBMERGED STRUCTURES

GENERIC "DRAG LOAD" METHODOLOGY AND PLANT UNIQUE FLOW FIELDS ARE BEING USED FOR LOCA CHUGGING ~~AND C.O.~~ LOADS ON SUBMERGED STRUCTURES IN COMPLIANCE WITH ACCEPTANCE CRITERIA.

GENERIC METHODOLOGY IDENTIFIES THREE COMPONENTS OF FLOW INDUCED LOADS ON SUBMERGED STRUCTURES: ACCELERATION DEPENDENT AND VELOCITY SQUARE DEPENDENT IN-LINE LOADS, VELOCITY SQUARE DEPENDENT LIFT LOAD (NORMAL TO THE DIRECTION OF FLOW).

PLANT UNIQUE FLOW FIELDS ARE BEING DEFINED CONSISTENTLY WITH CHUGGING ~~AND C.O.~~ BOUNDARY LOADS.

REPRESENTATIVE PLANT UNIQUE CHUGGING FLOW FIELDS SHOW THAT THE CHUGGING LOADS ON SUBMERGED STRUCTURES ARE DUE TO ACCELERATION OR PRESSURE GRADIENTS ESTABLISHED IN THE POOL DURING THE IMPULSIVE CHUGGING PHENOMENON, I.E., VELOCITY DEPENDENT LOADS ARE SMALL.