EVALUATION

OF

REACTOR COOLANT SYSTEM LOADS

AND

COMPONENT SUPPORT MARGINS

RESULTING FROM

OPTIMIZED REACTOR COOLANT PUMP SUPPORT CONFIGURATION

CRYSTAL RIVER 3 GENERATING PLANT

Prepared for
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by

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Florida Power Corporation (FPC), Babcock & Wilcox (B&W) and the NRC staff have discussed on several occasions the application of advanced fracture mechanics techniques to certain postulated pipe breaks in the Reactor Coolant System (RCS) main loop piping. FPC, based on these discussions and information provided has proposed to utilize those techniques to eliminate mechanical and structural load effects associated with postulated RCS main loop pipe breaks. In Reference (1), the NRC indicated that advanced fracture mechanics could be employed as a basis for an alternate approach these postulated pipe breaks. FPC in Reference (2) submitted a request for partial exemption from General Design Criteria 4 (GDC-4).

Specifically in Reference (2), FPC requested a partial exemption from those portions which require protection of structures, systems, and components against certain dynamic (including mechanical and structural loading) effects associated with postulated RCS main loop pipe breaks. This exemption pertains to all postulated breaks specified in the Crystal River-3 RCL piping. FPC has not requested exemption from GDC-4 for other postulated breaks. The request does not affect the CR-3 Nuclear Generating Station design basis for environmental, containment, equipment qualification or ECCS analysis.

The following information is provided as additional justification for the exemption:

1. In Reference (3), Babcock and Wilcox submitted for NRC staff review a fracture mechanics analysis to validate the "Leak-Before-Break" (LBB) failure scenario for their Nuclear Steam System (NSS) designs. Staff review of this submittal is complete and no significant deficiencies have been identified. Approval is pending based on completion of materials testing and minor revisions.

2. Reference (3) demonstrates that for the NSS-177 FA RCS main loop piping:
   a) A substantial sized flaw in the piping would not grow through the wall nor extend significantly in length during the plant design lifetime.
   b) If a flaw were to grow through the wall of the piping, it would open sufficiently to leak many times in maximum allowable leakage before extending anywhere near critical crack length.
   c) A very long through wall crack (many times longer than a leak detectable longitudinal or circumferential crack length) would remain stable under normal operation plus SSE loadings.

This demonstration provides sufficient justification for elimination of large postulated breaks from the design basis for the CR-3 Unit's RCS main loop piping.

3. The ACRS in Reference 4 has approved the application of the aforementioned fracture mechanics techniques to the analysis of asymetric blowdown loads. Reference (4) states "That is, there is no known mechanism in PWR primary piping material for developing a large break without going through an extended period during which the crack would leak copiously."

4. Appendix A presents an assessment of the new RCS pump supports configuration loadings with respect to the loadings evaluated in the LBB Report, Reference 3. The new RCS pump support configuration was outlined in the GDC-4 exemption request, Reference 2.

   The new support configuration loadings are enveloped by the generic analysis performed in support of the LBB report. The generic loadings exhibit a margin of 6.6% over the new FPC loadings at the maximum load location.

At other locations, the margin is significantly larger.

5. Appendix B presents an assessment of the RCS Components Supports seismic margin using loadings from the new RCS pump support configuration analysis. The new support configuration margins are calculated using techniques employed in the LLNL probalistic evaluation of B&W plants, Reference 5. The new and old margins are listed along with the minimums margin of any B&W plant evaluated.

In conclusion, there would be no adverse effect on safety resulting from the exemption. If the exemption is granted, it would have no effect on the potential for occurrence or the severity of accidents previously considered by the staff.

References

1. Generic Letter 84-04, D. G. Eisenhut to PWR Licensees, Construction Permit Holders and Applicants for Construction Permits, dated February 1, 1984.
2. Letter 3F0285-02, G. R. Westafer to H. R. Denton, "Request for Exemption From a Portion of 10CFR50, Appendix A, General Design Criteria 4" dated February 1, 1985.
3. Babcock and Wilcox Owners Group Report, "Leak-Before-Break Evaluation of Margins Against Full Break for RCS Primary Piping of B&W Designed NSS," B&W Topical Report BAW-1847, dated September 1984.
4. ACRS Letter, J. J. Ray to W. J. Dircks, "Fracture Mechanics Approach to Pipe Failure," dated June 14, 1983.
5. Ravindra, M. K. et. al., Probability of Pipe Failure in the Reactor Coolant Loop of Babcock and Wilcox PWR Plants, Volume 2: Guillotine Break Indirectly Induced by Earthquakes, Lawrence Livermore National Laboratory, UCRL-53644, NUREG/CR-4290, Vol. 2, (1985).

Appendix A

The LBB evaluation for B&W nuclear plants was performed using piping loads at various locations on the hot and cold legs of the RCS piping. These loadings were obtained from existing stress reports for each of the B&W Owners Group plants. From these loads a single set of generic loads was evaluated. Tables 4-1 and 4-6 of Reference 3 list the controlling generic load sets. Other load sets were evaluated; but these are limiting. The next page is a listing of the Crystal River 3 piping loadings resulting from the reconfiguration of the RC pump supports vers s the generic load set evaluated in the LBB report. The loadings are considered to be enveloped if at the limiting locations the new Crystal River 3 maximum is less and the new Crystal River 3 minimum is greater than those loading evaluated in the LBB report.

It can be seen that the moments resulting from the reconfiguration of the RC pump supports are within the envelope of moments that have been justified relative to Leak Before Break criteria. Therefore, the moments resulting from the reconfiguration of the RC pump supports are also acceptable for the Leak Before Break criteria.

CRYSTAL RIVER VERSUS GENERIC EVALUATION:

PIPE SIZE	GENERIC MAXIMUM MOMENT (FT-KIPS)	FPC MAXIMUM MOMENT (FT-KIPS)
28" I.D.		
STRAIGHT	3098.0 AT Jt. 12	2004.4 AT Jt. 21.5
ELBOW	2822.8 AT Jt. 24	1617.0 AT Jt. 27.5
36" I.D.		
STRAIGHT	2376.5 AT Jt. 9	2230.1 AT Jt. 9
ELBOW	2376.5 AT Jt. 9	2230.1 AT Jt. 9

PIPE SIZE	GENERIC MINIMUM MOMENT (FT-KIPS)	FPC MINIMUM MOMENT (FT-KIPS)
28" I.D.		
STRAIGHT	560 AT Jt. 12	1556.1 AT Jt. 21.5
ELBOW	1246 AT Jt. 24	1294.3 AT Jt. 27.5
36" I.D.		
STRAIGHT	1010 AT Jt. 9	1738.0 AT Jt. 9
ELBOW	1010 AT Jt. 9	1738.0 AT Jt. 9

Appendix B

The requirement to design the Crystal River - 3 Nuclear Power Plant (CR-3) for the effects of an instantaneous double-ended guillotine break (DEGB) of the reactor coolant loop (RCL) piping has led to excessive design costs, interference with normal plant operation and maintenance, and unnecessary radiation exposure of plant maintenance personnel. NRC/Lawrence Livermore National Laboratory (LLNL) sponsored a research program aimed at exploring whether the probability of DEGB in RCL Piping of nuclear power plants is acceptability small and if the requirements to design for DEGB effects (e.g., provision of pipe whip restraints) may be removed. Reference 5 describes the study performed for ten Babcock & Wilcox (B&W) plants of which CR-3 was included. Reference 5 estimates the probability of indirect DEGB in RCL piping as a consequence of seismic-induced structural failures within the containment of B&W supplied pressurized water reactor nuclear power plants in the United States.

The Reference 5 study used actual component support loadings to determine the seismic capacity factors for the component supports. Below is a listing of the new capacity factors (calculated in a manner similar to LLNL methodogy) for CR-3 and the factors calculated in the LLNL study for CR-3 and the minimum calculated for any B&W plant.

Capacity Factors of Safety for CR-3

	New	Old	Min. B&W
Reactor Pressure Vessel	12.7	5.56	3.76
Steam Generator	34.2	118.8	2.57
Reactor Coolant Pump	4.6	N/A	9.91

Only the strength factor F_S was recalculated in the above. The probabity of indirect DEGB at CR-3 due to new capacity factors and methods of analysis was not recalculated.

However, from Figure 2-8 of Reference 5 it can be inferred that a 50% reduction in capacity would result in a probability reduction of approximately one order of magnitude. Therefore, the new support configuration at CR-3 has more than sufficient margin when compared to PWR nuclear plants in total.

ATTACHMENT 1

Description of Computer Codes STALUM and RESPECT

Program Name	Purpose	Description	Classification
STALUM	General static, thermal, and dynamic analysis for linear elastic and gap structures	STALUM analyzes three-dimensional, finite segment systems consisting of uniform or nonuniform piping segments, closed-loop arrangements, and supporting elements. STALUM performs both static and dynamic structural analyses undergoing small linear, elastic deformations. The static analysis is based on the matrix displacement method. The static loadings are static mechanical forces, thermal, and/or support displacement loadings. The dynamic analysis is based on lumped mass and normal mode extraction techniques. The dynamic input loadings can be a response spectra or force time history. The essential input to the program consists of the physical properties of the system, the boundary conditions, and/or the loading information; the essential output consists of the resultant point displacement, rotations, forces, and moments at both ends of each segment and stresses at various locations in each segment.	B&W proprietary
RESPECT	To calculate amplified response spectra	RESPECT calculates the maximum acceleration response of a single degree of freedom (SDOF) oscillator subjected to an input acceleration time history at the base. An SDOF is described by a second order differential equation which contains a coefficient described as the Eigen value or natural frequency squared. When this equation is solved for input acceleration time history with varying Eigen values, the resulting maximum acceleration response and natural frequency form an acceleration response spectra (ARS). This program also calculates an ARS due to structural amplification between a known point and an attachment point. This technique requires a structural response spectra solution (accelerations and inertia forces) and the associated acceleration time history.	B&W proprietary

ATTACHMENT 2

OUTLINE OF METHODOLOGY USED FOR CR-3 PLANT SPECIFIC SEISMIC AND THERMAL ANALYSIS WITH ALTERNATE SUPPORT CONFIGURATION PERMITTED BY LBB CONCEPT

- Developed primary loop engineering mathematical model for seismic and thermal analysis. Revised an existing model to reflect CR-3 design for component support locations, stiffness(es), configuration, shield wall design and thermal properties. Seismic and thermal analyses were performed by B&W with their computer program STALUM; a fully certified, finite element piping analysis program designed for seismic applications via the response spectrum technique.

- Performed deadweight, thermal and Operating Basis Earthquake (OBE) seismic analysis for the revised support configuration, using increased damping.

- The combination of structural vibration modes was by the closely spaced modes technique as described in NRC Regulatory Guide 1.92.

- Amplified response spectra (for attachments to the RCS piping and pumps) were generated by B&W with their computer program RESPECT, which is a fully certified analysis program which calculates acceleration response spectra due to structural vibration amplification between a known point and an attached point.

- Revised nozzle spectra were compared to existing design spectra for spray, HPI, drain and letdown lines.

- Assured that all new piping loads were bounded by the loads utilized in the B&W Owners Group LBB Evaluation, BAW 1847.

- Damping considerations included:

  Original damping for components, shield walls, and vital piping systems is 0.5% of critical damping (as listed in FSAR and Reg. Guide 1.61); RCS analysis for optimized support configuration utilizes variable damping from 2-5% of critical damping for piping per ASME Code Case N-411 (a value of 2% was utilized for components and 4% for shield walls using Regulatory Guide 1.61).

- Combination of directional earthquake response was per the absolute sum method-maximum of x + y or y + z earthquakes (FSAR 5.4.5.2); pump loadings were combined by SRSS of the x, y, z direction earthquake loads.

- Design codes utilized:

  Power Piping; USAS B31.1.0 (1967) with erratum dated March 1969. This is the original construction code for the existing RCS pump supports.

Nuclear Power Piping; USAS B31.7 (February 1968 with erratum dated June 1968). This is the original construction code for the existing RCS piping.

ASME Section III, Class A (1965 Edition with Addenda through Summer 1967). This is the original construction code for the existing RCS pressure vessels.

ASME Nuclear Code Case N-411 (September 17, 1984).

ATTACHMENT 3

DESCRIPTION OF SNUBBER ASSEMBLIES, OVERALL SNUBBER CONFIGURATION, AND PUMP SUPPORT CONFIGURATION PLANNED FOR CR-3 ALTERNATE SUPPORT CONFIGURATION PERMITTED BY LBB CONCEPT

SNUBBER ASSEMBLY
DESCRIPTION

RC PUMP 3A1

Tag No.	Type	Assembly Length (inches)	Estimated Weight (pounds)
RCHS -1A	2000K	64.58	7320
RCHS-2A	1200K	77.68	4454
RCHS-3A	2000K	72.82	7468
RCHS-4A	1200K	89.38	4620
RCHS-5A	1000K	151.38	4455
RCHS-6A	1000K	116.38	4016
RCHS-7A	1200K	123.31	5104
RCHS-8A	1000K	54.50	3242

RC PUMP 3A2

Tag No.	Type	Assembly Length (inches)	Estimated Weight (pounds)
RCHS-1B	1600K	61.00	5471
RCHS-2B	1600K	63.12	5505
RCHS-3B	2000K	68.78	7396
RCHS-4B	1000K	216.00	5815
RCHS-5B	2000K	167.34	9171
RCHS-6B	1000K	185.50	4882
RCHS-7B	1200K	260.75	8588
RCHS-8B	1600K	211.12	8441

SNUBBER ASSEMBLY
DESCRIPTION

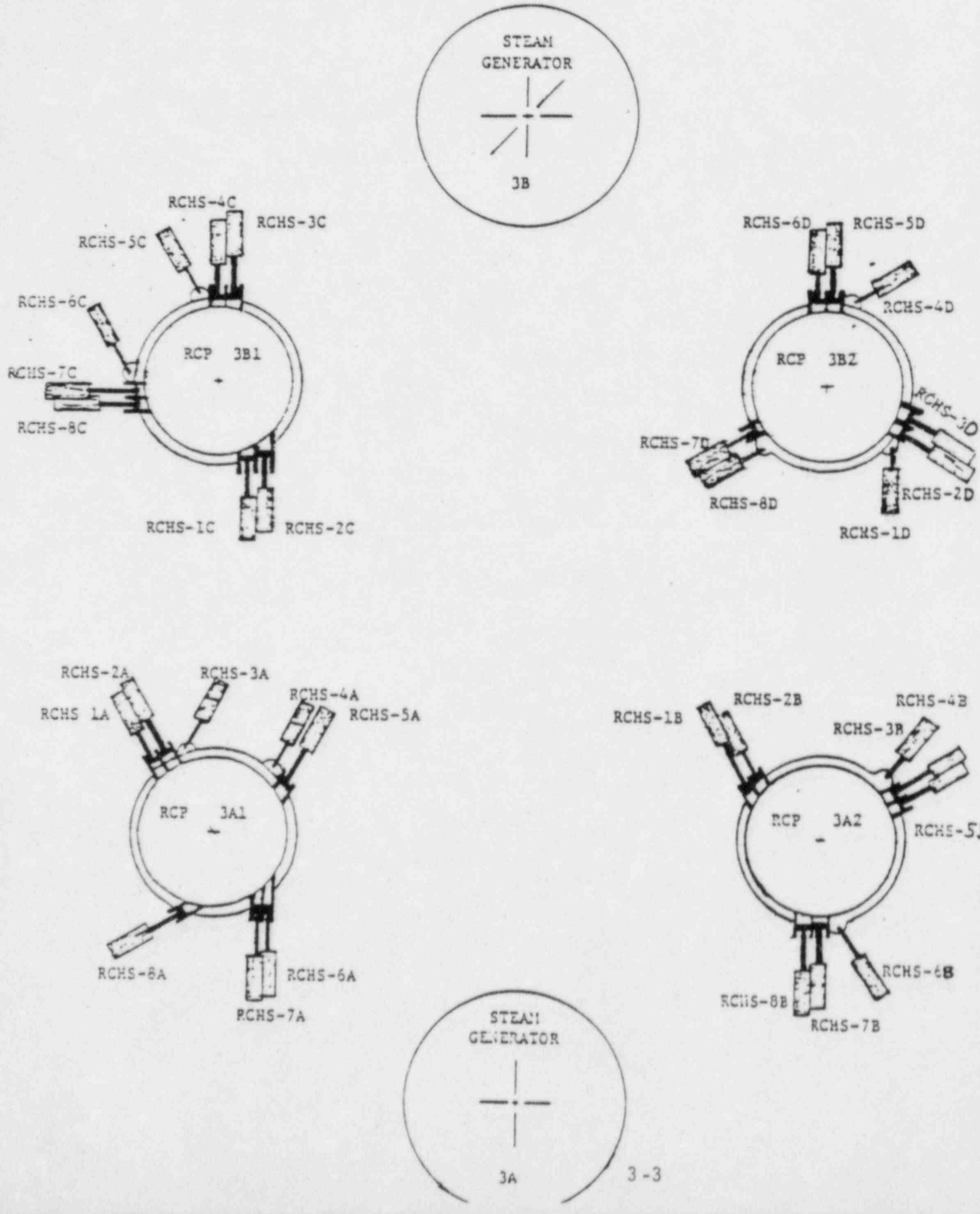
RC PUMP 3B1

Tag No.	Type	Assembly Length (inches)	Estimated Weight (pounds)
RCHS-1C	1200K	57.75	4140
RCHS-2C	1000K	58.18	3288
RCHS-3C	1000K	134.25	4241
RCHS-4C	1600K	155.75	6995
RCHS-5C	1200K	98.04	4744
RCHS-6C	2000K	99.50	7949
RCHS-7C	1000K	117.50	4031
RCHS-8C	1600K	102.43	6138

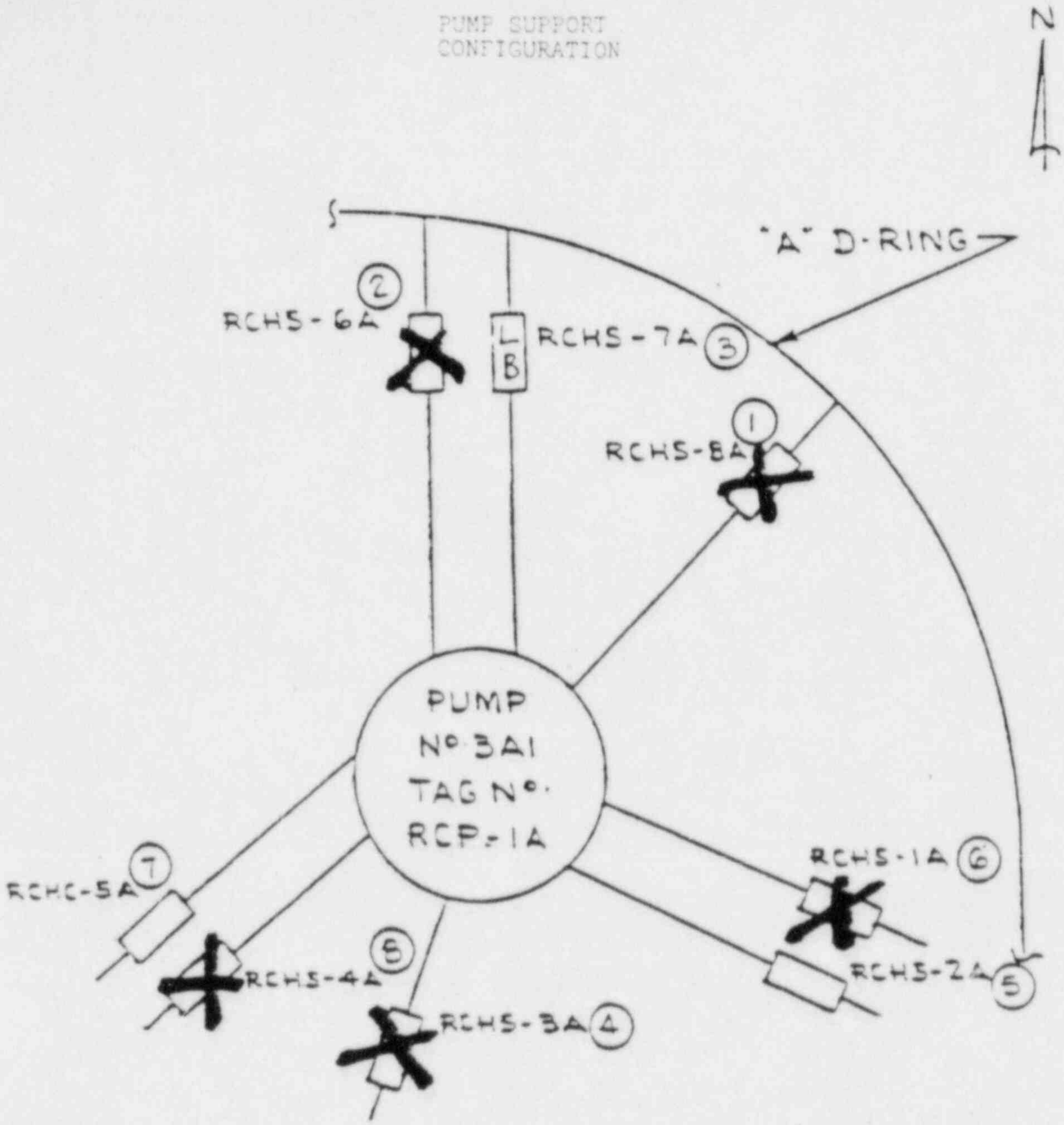
RC PUMP 3B2

Tag No.	Type	Assembly Length (inches)	Estimated Weight (pounds)
RCHS-1D	2000K	79.86	7595
RCHS-2D	1600K	117.18	6375
RCHS-3D	1000K	126.75	4147
RCHS-4D	1600K	65.68	5547
RCHS-5D	1600K	103.75	6159
RCHS-6D	2000K	95.78	7882
RCHS-7D	1000K	160.00	4563
RCHS-8D	1000K	101.00	3824

OVERALL SNUBBER
CONFIGURATION



PUMP SUPPORT
CONFIGURATION

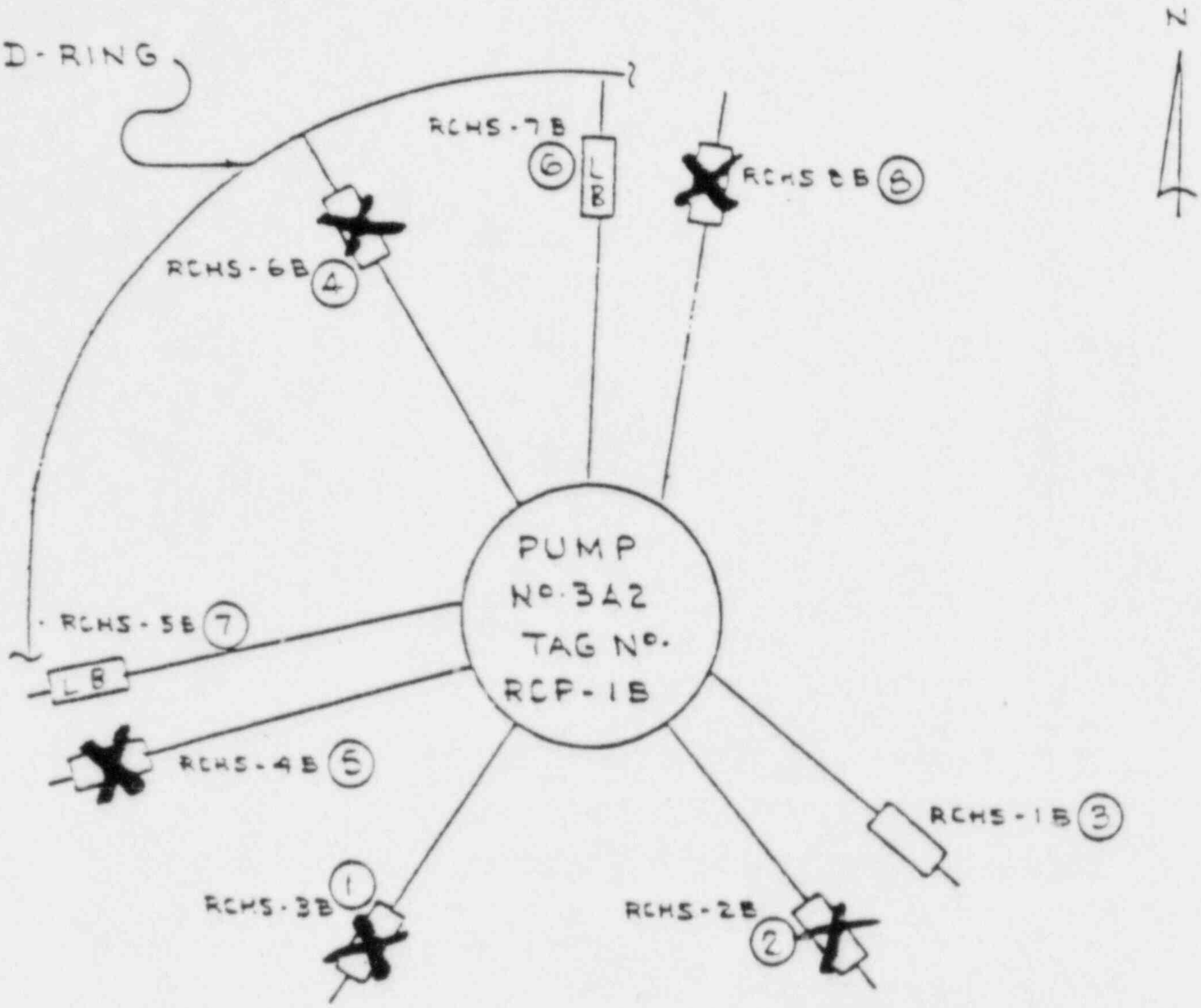


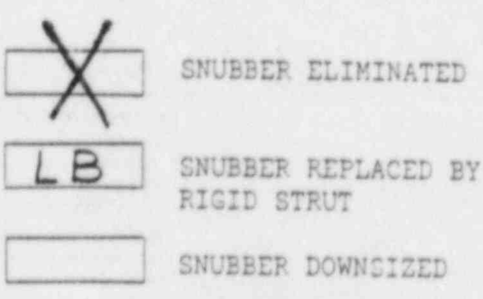
SNUBBER ELIMINATED

LB SNUBBER REPLACED BY RIGID STRUT

SNUBBER DOWNSIZED

PUMP SUPPORT
CONFIGURATION





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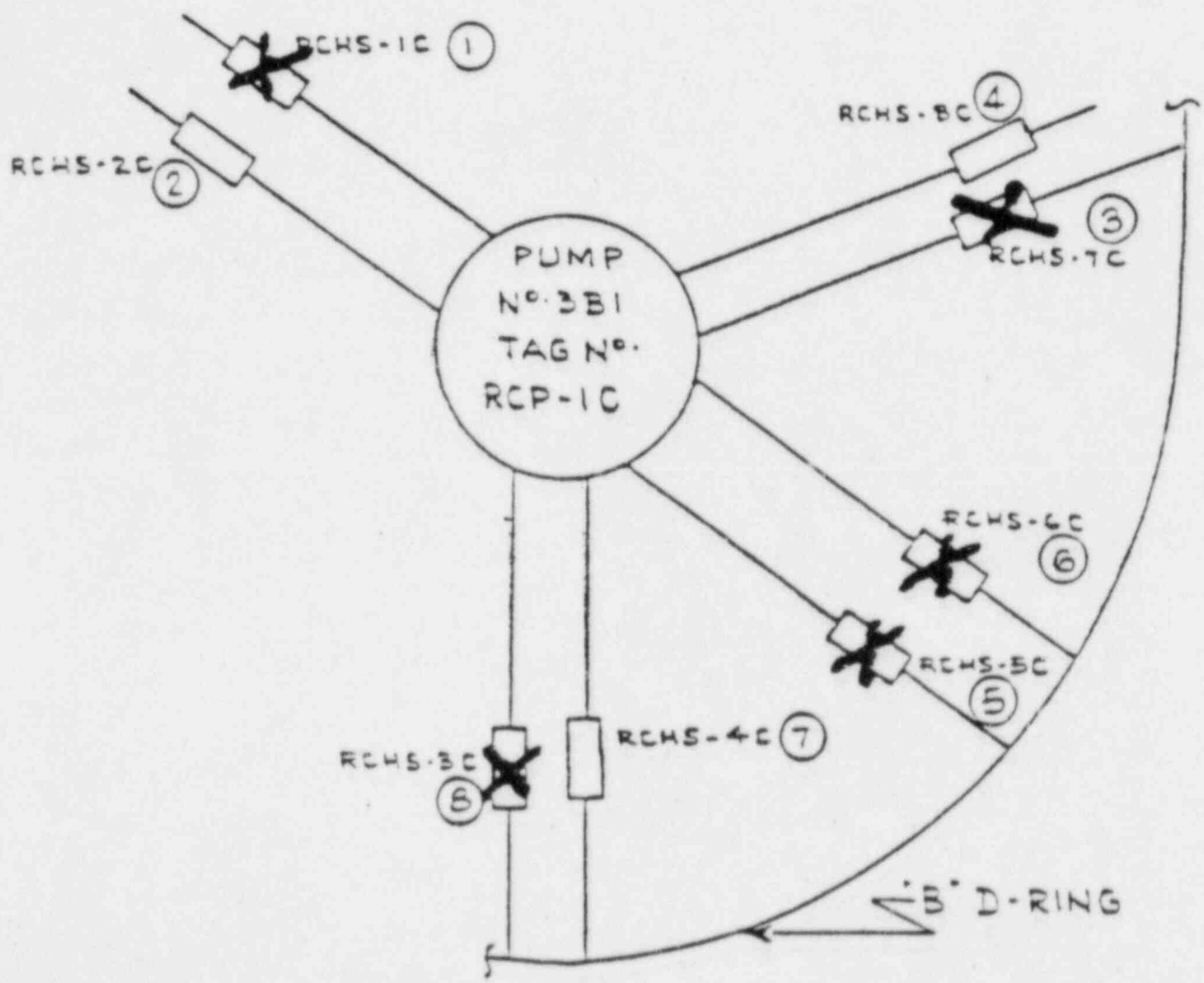

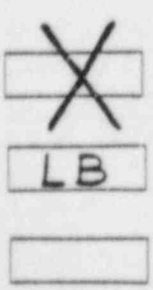

SNUBBER ELIMINATED

SNUBBER REPLACED BY
RIGID STRUT

SNUBBER DOWNSIZED

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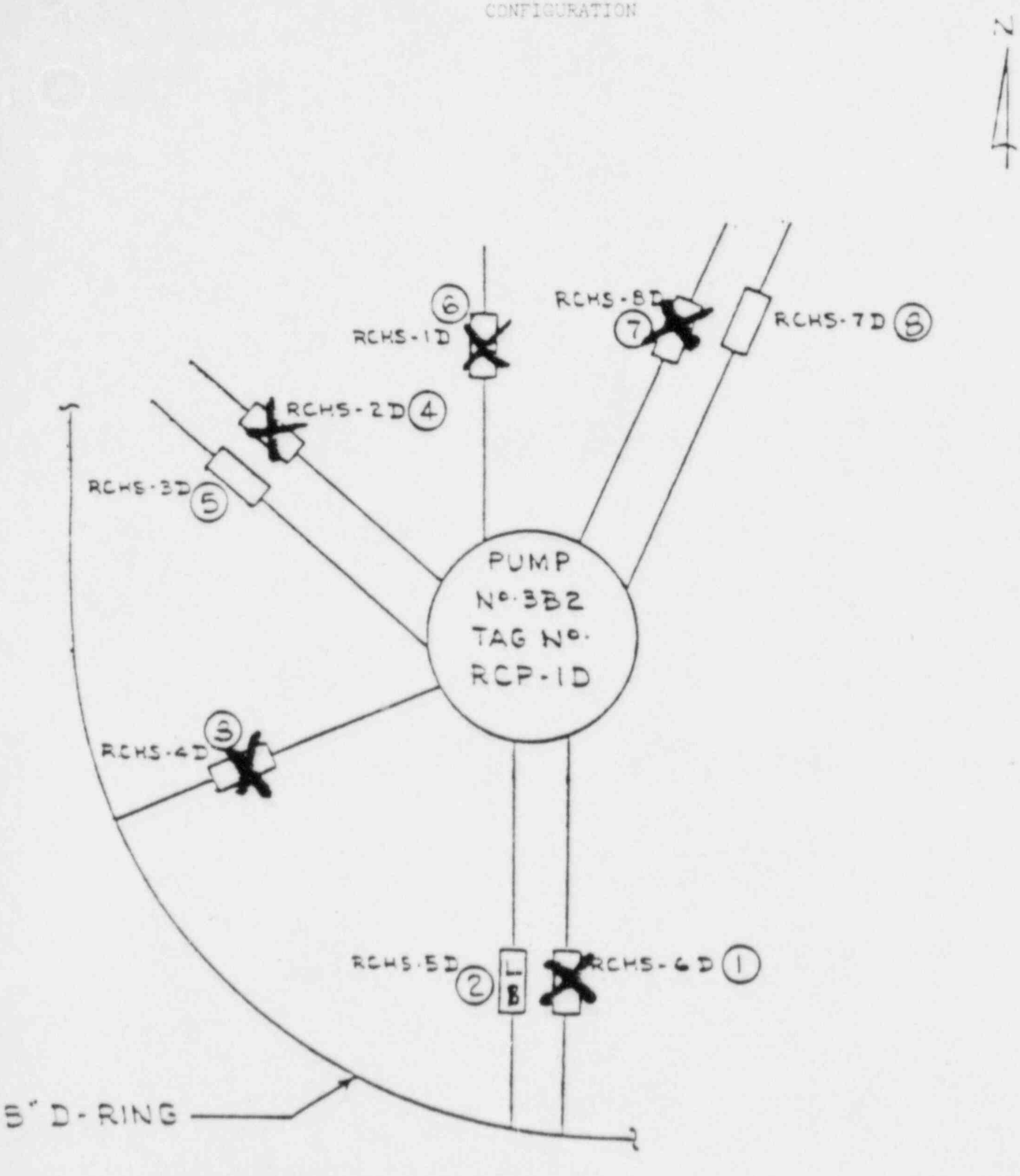
CONFIGURATION



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