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RECIP.NAME	RECIPIENT AFFILIATION

SUBJECT: LER 93-028-00:on 930923,identified unanalyzed HELB in primary containment.Caused by less than adequate design analysis & review of design analysis by architect engineer. Review will be performed of other piping sys.W/931025 ltr.

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WASHINGTON PUBLIC POWER SUPPLY SYSTEM

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October 25, 1993
G02-93-258

Docket No. 50-397

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U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Subject: **NUCLEAR PLANT WNP-2, OPERATING LICENSE NPF-21**
LICENSEE EVENT REPORT NO. 93-028-00

Transmitted herewith is Licensee Event Report No. 93-028-00 for the WNP-2 Plant. This report is submitted in response to the report requirements of 10CFR50.73 and discusses the items of reportability, corrective action taken, and action taken to preclude recurrence.

If the corrective actions identified in the LER result in additional reportable information, the results will be documented as a supplement to this LER.

Sincerely,

J. V. Parrish (Mail Drop 1023)
Assistant Managing Director, Operations

JVP/KBL/lr
Enclosure

cc: Mr. B. H. Faulkenberry, NRC - Region V
Mr. R. Barr, NRC Resident Inspector (Mail Drop 927N, 2 Copies)
INPO Records Center - Atlanta, GA
Mr. D. L. Williams, BPA (Mail Drop 399)

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LICENSEE EVENT REPORT (LER)

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
Washington Nuclear Plant - Unit 2	0 5 0 0 0 3 9 7	1 OF 10

TITLE (4)
FAILURE TO ANALYZE A HIGH ENERGY LINE BREAK LOCATED OUTSIDE PRIMARY CONTAINMENT

EVENT DATE (5)			LER NUMBER (6)			REPORT DATE (7)			OTHER FACILITIES INVOLVED (8)	
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	MONTH	DAY	YEAR	FACILITY NAMES	DOCKET NUMBERS(S)
09	23	93	93	028	00	10	25	93		05000
										05000

OPERATING MODE (9): 1

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more of the following) (11)

POWER LEVEL (10): 100

☐ 20.402(b)	☐ 20.405(C)	☐ 50.73(a)(2)(iv)	☐ 77.71(b)
☐ 20.405(a)(1)(i)	☐ 50.36(c)(1)	☐ 50.73(a)(2)(v)	☐ 73.73(c)
☐ 20.405(a)(1)(ii)	☐ 50.36(c)(2)	☐ 50.73(a)(2)(vii)	☐ OTHER (Specify in Abstract below and in Text, NRC Form 366A)
☐ 20.405(a)(1)(iii)	☐ 50.73(a)(2)(i)	☐ 50.73(a)(2)(viii)(A)	
☐ 20.405(a)(1)(iv)	X 50.73(a)(2)(ii)	☐ 50.73(a)(2)(viii)(B)	
☐ 20.405(a)(1)(v)	☐ 50.73(a)(2)(iii)	☐ 50.73(a)(2)(x)	

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER
Kurt B. Lewis, Technical Specialist	AREA CODE 509 377-4145

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT (13)

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS	CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS

SUPPLEMENTAL REPORT EXPECTED (14)		EXPECTED SUBMISSION DATE (15)	MONTH	DAY	YEAR
☐ YES (If yes, complete EXPECTED SUBMISSION DATE)	X NO				

ABSTRACT (16)

On September 23, 1993, with the reactor in MODE 1 at 100% power, a programmatic engineering review of high energy line break (HELB) analysis to confirm consistency between Leak Detection (LD) system capabilities and HELB analysis assumptions identified an unanalyzed HELB. The WNP-2 Final Safety Analysis Report (FSAR), Section 3.6, indicates that HELBs were analyzed for environmental effects, specifically temperature and humidity. However, the programmatic engineering review determined that these environmental effects were not analyzed for a postulated break in a four-inch Reactor Water Cleanup (RWCU) system line RWCU(5)-3. The break is open to the Reactor Building environment.

Engineering immediately performed an operability evaluation for the unanalyzed HELB conditions and determined that the plant could continue to operate safely. Further corrective action consists of finalizing revisions to Supply System HELB analysis and requesting permanent exclusion of the postulated break.

The root cause of this event was less than adequate design analysis and review of design analysis by the architect engineer. There were no contributing causes to this event.

This event posed no threat to the health and safety of the public or plant personnel.

Plant Conditions

Power Level - 100%
Plant Mode - 1 (Power Operation)

Event Description

On September 23, 1993, with the reactor in MODE 1 at 100% power, a programmatic engineering review of high energy line break (HELB) analysis to confirm consistency between Leak Detection (LD) system capabilities and HELB analysis assumptions identified an unanalyzed HELB. The WNP-2 Final Safety Analysis Report (FSAR), Section 3.6, indicates that HELBs were analyzed for pipe whip, jet impingement, flooding, pressurization, and environmental effects, specifically temperature and humidity. However, the programmatic engineering review determined that environmental effects were not correctly analyzed for one HELB.

This HELB involves a postulated break in four-inch Reactor Water Cleanup (RWCU) system line RWCU(5)-3 (Figure 1). Specifically, this break is located at RWCU system blowdown flow control valve RWCU-FCV-33 and is open to Reactor Building floor elevation 501′. The postulated break location involves ASME Section III Class 3 piping. Postulated pipe break locations are based on guidelines provided by NRC Branch Technical Positions APCSB 3-1 and MEB 3-1 (as described in FSAR 3.6.2.1). In part, for ASME Section III Class 3 piping, these guidelines require breaks to be postulated at terminal ends. For piping runs which are maintained pressurized for only a portion of the run, MEB 3-1 defines a terminal end as the piping connection at the first normally closed valve in the run. Accordingly, a break should be postulated at RWCU-FCV-33. During power operation, this line operates at primary coolant pressure at an approximate temperature of 125 degrees Fahrenheit.

Immediate Corrective Action

Engineering performed an operability evaluation for the postulated HELB which included completing nondestructive examination testing of the postulated break location. The evaluation determined that the plant could continue to operate safely. The determination was based principally on the low stress levels at the postulated break locations, the positive results of the nondestructive examination testing, and diverse means to isolate the postulated RWCU line break automatically.

Further Evaluation, Root Cause, and Corrective Action

Further Evaluation

1. On September 23, 1993, at approximately 1357 hours, the failure to analyze two HELBs located outside Primary Containment was reported to the NRC by telephone in accordance with 10CFR50.72(b)(1)(ii)(B) which requires the Licensee to notify the NRC within one hour of conditions outside the design bases of the plant. A further evaluation of this event concluded that more appropriate reporting criteria were 10CFR50.72(b)(1)(ii)(A) and 50.73(a)(2)(ii)(A), "...an unanalyzed condition that significantly compromises(ed) plant safety." Continued review showed only the postulated RWCU HELB met these reportability criteria. Further evaluations of the analysis associated with the second break involving the postulated Heating Steam Condensate (HCO) HELB concluded this condition was not reportable.

2. There were no structures, components, or systems inoperable prior to this event that contributed to the event.

3. As stated, on September 23, 1993, a WNP-2 engineer discovered an unanalyzed HELB while performing the programmatic review as follows:

   a. The review was performed in response to General Electric Potentially Reportable Condition Report PRC 88-17, "Main Steam Tunnel Temperature Instrumentation and Isolation", issued in May 1989. This report consisted of three recommended actions. The first recommendation consisted of reviewing coverage of exhaust gas radiation monitoring capability. The second recommendation consisted of reviewing reactor coolant pressure boundary leak detection/isolation initiation to confirm consistent application of the Leak Detection system design intent. These two recommendations were completed in January 1990 and March 1992, respectively.

   b. The third recommendation advised utilities to review "HELB analysis assumptions ... for installed LD system capability." The purpose of this recommendation was to identify potential oversights in the capability of the WNP-2 leak Detection system in light of information provided in the PRC 88-17 report. Preliminary review of this recommended action, as well as similar reviews performed prior to this recommendation did not discover any significant problems. Since this recommendation required extensive review for closure, the review competed with other priority tasks and resulted in an extended schedule for completion. The review actually involved evaluation of over 160 break locations and 100 leak detection sensors. The postulated RWCU HELB was discovered near the end of the review.

4. Upon discovery of the unanalyzed HELB, engineering initiated an investigation to determine why the HELB was not evaluated. The investigation determined that the list of bounding HELBs used to analyze corresponding environmental effects was developed from an earlier list of bounding HELBs that omitted the postulated break in line RWCU(5)-3.

Root Cause

The root cause of this event was less than adequate design analysis and design analysis review by the architect engineer. The programmatic review performed by the Supply System determined that the architect engineer omitted the RWCU HELB from the list of bounding HELBs because the effects of the associated blowdown were incorrectly assessed. The architect engineer erroneously concluded that because water contained within this particular branch of piping was of moderate temperature, a postulated break at this location would have negligible environmental impact. However, the postulated RWCU HELB would cause a rapid increase in blowdown temperature until blowdown was terminated. The rapid increase in temperature would occur because the postulated break would significantly reduce the amount of water returning to the reactor via the shell-side of the regenerative heat exchanger RWCU-HX-1A(B,C). This would have the effect of removing the heat sink for water at reactor temperature entering the tube-side of the regenerative heat exchanger and thus causing a rapid increase in blowdown temperature. Because this list later formed the list of bounding HELBs subsequently analyzed for environmental effects, the mistake was carried forward to that analysis. Prior to initial plant startup, the Supply System overview of the architect engineer's work was not to the level of detail where this error would be discovered. There were no contributing causes to this event.

Further Corrective Action

1. A review will be performed to ensure that other piping systems were not inadvertently omitted from consideration of HELB environmental effects. This review will evaluate the other piping systems eliminated from consideration by the architect engineer. This will be completed by June 1, 1994.

2. A request for permanent exclusion of the postulated RWCU(5)-3 line break based on the as-built stress analysis and recent nondestructive examination results will be submitted to the NRC by December 3, 1993.

Safety Significance

Introduction

The RWCU HELB is safety significant in that design basis environmental conditions in the Reactor Building would be exceeded if no automatic isolation were credited. However, the Supply System believes continued plant operation is justified based on the low stress levels associated with the postulated break, the results of recent nondestructive examination testing performed at the postulated break, and the diverse mitigation features equipment capable of isolating the postulated break if it occurred. Following is a discussion of the environmental effects, stress analysis, and mitigating features associated with the postulated RWCU HELB.

Environmental Effect

If no automatic isolation of the RWCU HELB were to occur, the environmental conditions utilized for qualifying various safety-related pieces of equipment in the Reactor Building would be exceeded. The affected area could be large because open Reactor Building equipment access hatches could allow wide dispersement of the blowdown environment.

Stress Analysis

NRC Branch Technical Positions APCSB 3-1 and MEB 3-1 require breaks to be postulated at terminal ends. For piping runs which are maintained pressurized for only a portion of the run, MEB 3-1 defines terminal end as the piping connection at the first normally closed valve in the run. Accordingly, a break should be postulated at RWCU-FCV-33. However, review of as-built stress analysis for the RWCU(5)-3 piping connection to RWCU-FCV-33 determined that calculated stresses due to various corresponding loading conditions are well below the ASME allowable values necessary to credibly postulate a HELB or a through-wall pipe crack:

ASME Sec. III, Class 3 Piping Stress Eq.	Analyzed Stress Effect	Calculated Value	ASME Allowable Value
Eq. 8	Drywell Temperature (DWT) + Pressure	5,483 psi	15,000 psi
Eq. 9	DWT + Pressure + Operating Basis Earthquake (OBE)	9,838 psi	18,000 psi
Eq. 10	Thermal Stress	360 psi	22,500 psi

Stress-based pipe breaks and cracks are required to be postulated when the summation of ASME Equations 9 and 10 exceed a specified portion of the ASME Code stress allowable values (FSAR Sections 3.6.2.1.1.2 and 3.6.2.1.3). The sum of ASME Equations 9 and 10 and the FSAR break and crack criteria are tabulated as follows:

Summation of Calculated Values for ASME Eq. 9 and 10	FSAR Stress Criteria for Full Guillotine Breaks	FSAR Stress Criteria for Through-Wall Cracks
10,198 psi	>/= 32,400 psi	>/= 16,200 psi

In addition to the calculated stress values, nondestructive examination of the subject RWCU piping was performed on September 23, 1993, under Maintenance Work Request (MWR) AP5406. The results of this examination revealed no significant degradation or erosion/corrosion of the subject piping from past plant operation.

Mitigating Features

There are three automatic isolation functions of the RWCU system not previously credited that will mitigate the consequences of the postulated RWCU HELB. These functions are the non-regenerative heat exchanger RWCU-HX-2A(B) high outlet temperature isolation, filter-demineralizer RWCU-DM-1A(B) high differential pressure isolation, and filter-demineralizer outlet resin trap RWCU-RST-70A(B) high differential pressure isolation. The high temperature isolation function closes Primary Containment isolation valve RWCU-V-4 which isolates the flow of reactor coolant to the inlet of the RWCU system. The RWCU-V-4 valve is fully qualified and is designed to isolate a HELB in the RWCU system. The filter-demineralizer high differential pressure isolation and the filter-demineralizer outlet resin trap high differential pressure isolation each individually effects filter-demineralizer isolation by closure of filter-demineralizer inlet valve RWCU-V-206A(B) and filter-demineralizer outlet flow control valve RWCU-V-266A(B). Although the three isolation functions are initiated by unqualified instruments and, with the exception of RWCU-V-4, are accomplished by unqualified valves, the Supply System believes these functions would provide reliable means of isolating the postulated RWCU HELB. None of the unqualified valves or instrumentation credited for isolating the postulated HELB would be affected by the postulated break because this equipment is either located outside the Reactor Building or in areas unaffected by the break consequences. Additionally, the Supply System reviewed the maintenance history of unqualified equipment including check valve RWCU-V-39 (isolates RWCU system return piping to the reactor), RWCU-V-206A(B), and RWCU-V-266A(B). The review indicated that the valves have operated reliably. Further, the Supply System verified with the "206" and "266" valve vendor that these valves are capable of closing under the evaluated flow and differential pressure values associated with the postulated RWCU HELB. Lastly, the Supply System determined that the unqualified instrument sensors generating the referenced isolation functions have been recently calibrated.

If the postulated break were to occur, the break would cause high flow in the RWCU system, which in turn would cause the temperature of the water exiting the non-regenerative heat exchangers to increase. An associated temperature element (RWCU-TE-7) and temperature indicating switch (RWCU-TIS-8) would sense the rise in water temperature and would initiate automatic closure of RWCU-V-4 and subsequent trip of the RWCU system pumps (RWCU-P-1A(1B)). The postulated RWCU HELB would be completely isolated, as RWCU-V-4 would isolate RWCU system inlet piping from the reactor, and check valve RWCU-V-39 would isolate RWCU system return piping to the reactor. The Supply System is confident that this particular isolation logic channel would respond as designed, as the Supply System has confirmed that the isolation function's Logic System Functional Test (surveillance procedure 7.4.3.2.2.16) was recently performed per schedule in June 1993 with successful response of the logic channel. This LSFT is of the same rigor as LSFTs associated with Class I components.

Our analysis indicates that RWCU system break flow would approach 2000 gpm within one second of the postulated break. This flow rate would cause the filter-demineralizer RWCU-DM-1A(B) high differential pressure setpoint to be exceeded which would initiate automatic closure of filter-demineralizer outlet flow control valve RWCU-FCV-266A(B) and, by interlock, closure of filter-demineralizer inlet valve RWCU-V-206A(B). Similarly, break flow would cause the filter-demineralizer outlet resin trap RWCU-RST-70A(B) differential pressure isolation setpoint to be exceeded which also causes automatic closure of RWCU-FCV-266A(B) and RWCU-V-206A(B). The "206" and "266" valves are air-operated, fail-closed on loss-of-air and/or power and are spring-loaded to close.

To evaluate the consequences of a postulated line break in the RWCU piping system, a conservative evaluation was performed based on the expected response of the previously referenced isolations. As stated, if the postulated break were to occur, high temperature in the RWCU system would initiate automatic closure of RWCU-V-4 and a subsequent trip of the RWCU pumps, as well as automatic closure of RWCU-V-266A(B) and RWCU-V-206A(B) due to high differential pressure across the associated filter-demineralizers and/or the filter-demineralizer resin traps. For conservatism, it was assumed that only RWCU-V-4 would close to mitigate the break. No credit for the quicker acting RWCU-V-266A(B) valves which would isolate on high differential pressure across the filter demineralizers and/or the associated resin traps, is assumed in the analysis. Crediting RWCU-V-4 closure results in a more limiting system response time and associated environmental impact. The evaluation included the effects of high temperature water (440 degrees Fahrenheit) in the regenerative heat exchangers expanding and flashing into steam at the 501′ elevation along with the pressurized lower temperature water (140 degrees Fahrenheit) on the filter-demineralizer side of the break. Results of the analysis indicated that peak temperature on the 501′ elevation would not exceed 160 degrees Fahrenheit and would drop below 120 degrees Fahrenheit within 600 seconds; humidity would peak at 100%. (Safety-related equipment in Reactor Building harsh environment areas has been evaluated and has been determined to be qualified for the environmental conditions of the event. The details of the above evaluation have been documented and filed with the plant problem report.

<u>Similar Events</u>

LER 85-001-01 identified nonconservative engineering assumptions associated with RWCU and Reactor Core Isolation Cooling (RCIC) system HELB calculations. The calculations were used in developing Reactor Building environmental profiles which in turn were used in determining equipment qualification. The Supply System made the discovery during in-house reviews of associated calculations performed by an independent contractor. Corrective actions associated with this event were focused on modifying valve motor operators to decrease the severity of the environmental profile in the Reactor Building.

<u>EIIS Information</u>

Text Reference	EIIS Reference System	EIIS Reference Component
Leak Detection System (LD)	IJ	--
Reactor Water Cleanup System (RWCU)	CE	--
Blowdown Flow Control Valve RWCU-FCV-33	CE	FCV
Primary Containment	NH	--
Heating Steam Condensate (HCO) (part of Heating Steam (HS) System)	--	--
Main Steam Tunnel	SB	--
Regenerative Heat Exchanger RWCU-HX-1A(B,C)	CE	HX
Drywell	NH	--
Non-Regenerative Heat Exchanger RWCU-HX-2A(B)	CE	HX
Filter-Demineralizer RWCU-DM-1A(B)	CE	FDM
Resin Trap RWCU-RST-70A(B)	CE	TRP
Primary Containment Isolation Valve RWCU-V-4	CE	ISV
Check Valve RWCU-V-39	CE	V
Inlet Valve RWCU-V-206A(B)	CE	ISV
Outlet Flow Control Valve RWCU-FCV-266A(B)	CE	FCV
Temperature Element RWCU-TE-7	CE	TE
Temperature Indicating Switch RWCU-TIS-8	CE	TIS
RWCU System Pumps RWCU-P-1A(1B)	CE	P
Reactor Core Isolation Cooling System (RCIC)	BN	--

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TITLE (4)
FAILURE TO ANALYZE A HIGH ENERGY LINE BREAK LOCATED OUTSIDE PRIMARY CONTAINMENT

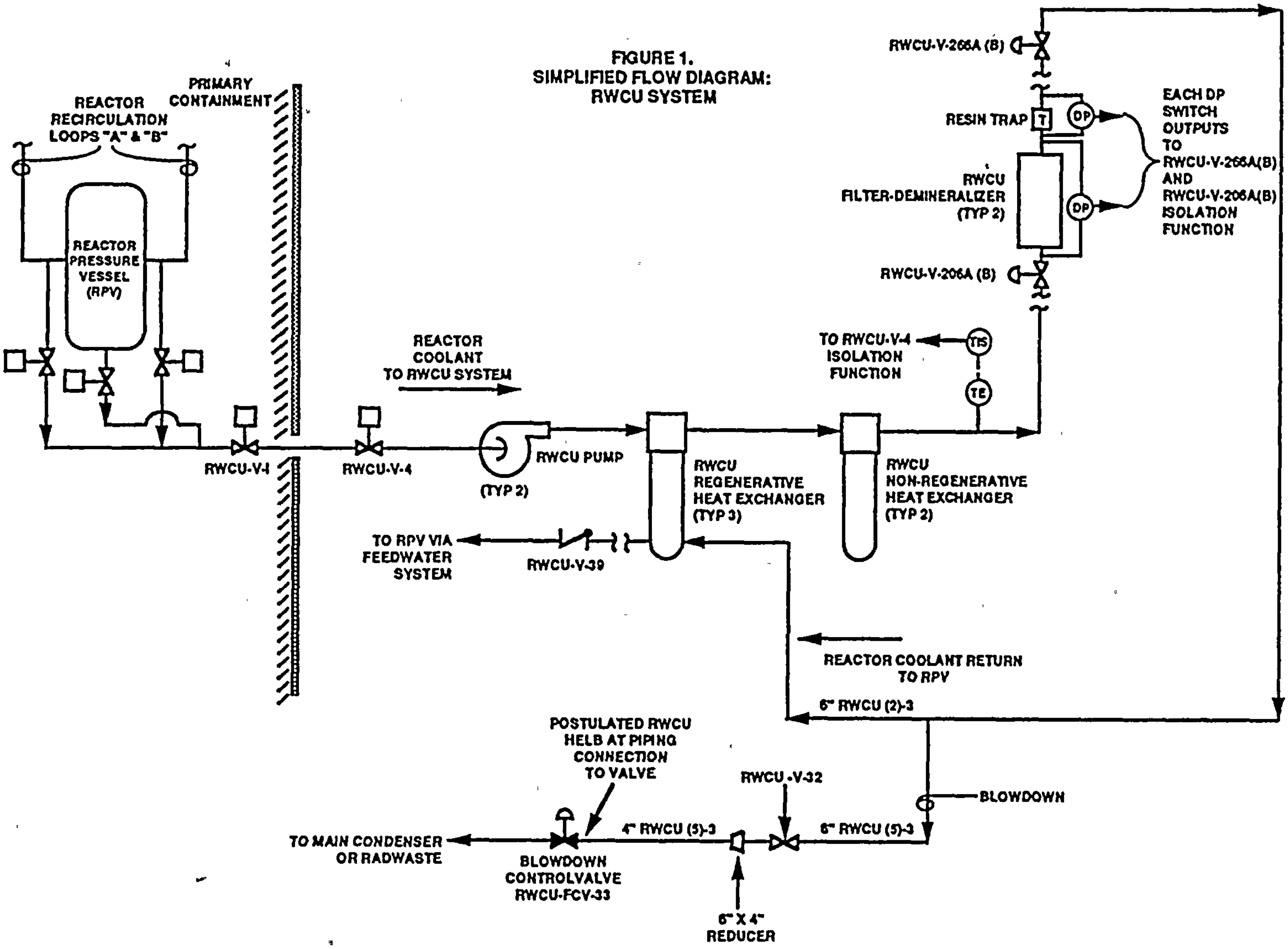

FIGURE 1.
SIMPLIFIED FLOW DIAGRAM:
RWCU SYSTEM