U.S. NUCLEAR REGULATORY COMMISSION
APPROVED OMB NO. 3150-0104
EXPIRES 8/31/85

LICENSEE EVENT REPORT (LER)

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
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TITLE (4)
INCORRECT CALIBRATION OF RESIDUAL HEAT REMOVAL FLOW INSTRUMENTATION

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 NUMBER(S)
0 7	0 9	8 5	8 5	– 0 3 1	– 0 1	0 9	0 9	8 5	D. C. COOK - UNIT 2	0 5 0 0 0 3 1 6
										0 5 0 0 0

OPERATING MODE (9): NA
POWER LEVEL (10): 0 0 0

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

20.402(b)	20.405(c)	50.73(a)(2)(iv)	73.71(b)
20.405(a)(1)(i)	50.36(c)(1)	50.73(a)(2)(v)	73.71(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)	X 50.73(a)(2)(i)	50.73(a)(2)(viii)(A)	
20.405(a)(1)(iv)	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 (AREA CODE)
A. A. BLIND - ASSISTANT PLANT MANAGER	6 1 6 4 6 5 - 5 9 0 1

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

CAUSE	SYSTEM	COMPONENT	MANUFAC-TURER	REPORTABLE TO NPRDS	CAUSE	SYSTEM	COMPONENT	MANUFAC-TURER	REPORTABLE TO NPRDS

SUPPLEMENTAL REPORT EXPECTED (14)

		EXPECTED SUBMISSION DATE (15)	MONTH	DAY	YEAR
YES *(If yes, complete EXPECTED SUBMISSION DATE)*	X NO				

ABSTRACT *(Limit to 1400 spaces, i.e., approximately fifteen single-space typewritten lines)* (16)

THIS IS A SUPPLEMENTAL REPORT TO LER 315/85-031-00 DATED AUGUST 8, 1985.

ON JULY 9, 1985, AT 1130 HOURS (WITH UNIT 1 SHUTDOWN FOR REFUELING WITH NO FUEL IN THE CORE AND UNIT 2 AT 100 PERCENT REACTOR THERMAL POWER), IT WAS DETERMINED THAT RESIDUAL HEAT REMOVAL (RHR) SYSTEM LOCAL FLOW MEASUREMENTS DID NOT COINCIDE WITH PROCESS INSTRUMENTATION INSTALLED IN THE SAFETY INJECTION PATH. ON JULY 15, 1985, AT 1600 HOURS (UNIT 1 STATUS UNCHANGED, UNIT 2 AT 95 PERCENT REACTOR THERMAL POWER), A SIMILAR DISCREPANCY WAS DISCOVERED CONCERNING RHR COOLDOWN FLOW INDICATION.

INVESTIGATION REVEALED THAT THE PROCESS INSTRUMENTATION WAS INDICATING HIGHER FLOWS THAN ACTUALLY EXISTED. THE INSTRUMENTS IN QUESTION ARE USED TO ENSURE COMPLIANCE WITH TECHNICAL SPECIFICATIONS 3.1.1.3 AND 3.9.8.1. ADDITIONALLY, THE COOLDOWN FLOW INDICATION IS USED AS A SIGNAL SOURCE FOR AN RHR LOW-FLOW ALARM. THE RECALIBRATION OF THE INVOLVED INSTRUMENTS (IFI-311, 321 AND 335), USING CORRECTED CALIBRATION DATA, WAS COMPLETED ON AUGUST 19, 1985. A REVIEW OF SIMILAR INSTRUMENTATION HAS BEEN CONDUCTED, NO OTHER ERRONEOUS CALIBRATION DATA HAS BEEN IDENTIFIED.

A SAFETY ANALYSIS OF DEGRADED RESIDUAL HEAT REMOVAL SYSTEM FLOW HAS BEEN COMPLETED. MINIMUM FLOW RATES REQUIRED TO KEEP THE REACTOR WATER TEMPERATURE BELOW 200°F WERE CALCULATED FOR TWO SETS OF ENVIRONMENTAL CONDITIONS. THE ANALYSIS SHOWED THAT THE RESULTING REDUCTION IN FLOW DID NOT CONSTITUTE AN UNREVIEWED SAFETY QUESTION.

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TEXT (If more space is required, use additional NRC Form 366A's) (17)

THIS IS A SUPPLEMENTAL REPORT TO LER 315/85-031-00 DATED AUGUST 8, 1985. THE INFORMATION CONTAINED HEREIN IS CURRENTLY UNDERGOING CORPORATE REVIEW, ANY SIGNIFICANT CHANGES AS A RESULT OF THIS REVIEW WILL BE REPORTED.

ON JULY 9, 1985, AT 1130 HOURS (WITH UNIT 1 SHUTDOWN FOR REFUELING WITH NO FUEL IN THE CORE AND UNIT 2 AT 100 PERCENT REACTOR THERMAL POWER), IT WAS DETERMINED THAT RESIDUAL HEAT REMOVAL (RHR) (IEEE/BP) SYSTEM LOCAL FLOW MEASUREMENTS, OBTAINED WHILE PERFORMING MAINTENANCE ACTIVITIES, DID NOT COINCIDE WITH PROCESS INSTRUMENTATION INSTALLED IN THE SAFETY INJECTION PATH. ON JULY 15, 1985, AT 1600 HOURS (UNIT 1 STATUS UNCHANGED, UNIT 2 AT 95 PERCENT REACTOR THERMAL POWER), A SIMILAR DISCREPANCY WAS DISCOVERED CONCERNING RHR COOLDOWN FLOW INDICATION.

SUBSEQUENT INVESTIGATION REVEALED THAT THE PROCESS INSTRUMENTATION WAS INDICATING HIGHER FLOWS THAN ACTUALLY EXISTED. THE INSTRUMENTS IN QUESTION ARE USED TO ENSURE COMPLIANCE WITH TECHNICAL SPECIFICATIONS 3.1.1.3 AND 3.9.8.1, WHICH REQUIRE A MINIMUM FLOW OF 3000 GPM WHEN USING THE RHR SYSTEM. THIS ENSURES THAT ADEQUATE DECAY HEAT REMOVAL EXISTS AND THAT THERE WILL BE SUFFICIENT FLUID MIXING TO PREVENT BORON STRATIFICATION IN THE EVENT OF A BORON DILUTION ACCIDENT. ADDITIONALLY, THE COOLDOWN FLOW INDICATION IS USED AS A SIGNAL SOURCE FOR AN RHR LOW-FLOW ALARM. THIS ALARM WAS SET FOR A VALUE OF 2000 GPM AND WAS INTENDED TO PREVENT PUMP DAMAGE IN THE EVENT OF INADVERTENT CLOSURE OF THE SHUTOFF OR ISOLATION VALVES IN THE RHR PUMP SUCTION LINES. THE RECALIBRATION OF THE INVOLVED INSTRUMENTS (IFI-311, 321 AND 335), USING CORRECTED CALIBRATION DATA, WAS COMPLETED ON AUGUST 19, 1985. A REVIEW OF SIMILAR INSTRUMENTATION HAS BEEN CONDUCTED, NO OTHER ERRONIOUS CALIBRATION DATA HAS BEEN IDENTIFIED.

INVESTIGATION OF THE INSTRUMENTATION REVEALED THAT THREE FACTORS CONTRIBUTED TO THE FLOW MEASUREMENT DISCREPANCY: 1) THE SCALE GRADUATION MARKS ON THE INDICATORS WERE IN ERROR, 2) THE TRANSMITTER CALIBRATIONS WERE DONE TO AN ERRONEOUS RANGE, AND 3) THE DIFFERENTIAL PRESSURE VERSUS FLOW CONDITIONS HAD BEEN CALCULATED AT A WATER TEMPERATURE OF 400°F. THE MAJOR SOURCES OF THE DISCREPANCY WERE THE SCALING AND TRANSMITTER ERRORS. THE WATER TEMPERATURE INTRODUCED AN ADDITIONAL ERROR, BECAUSE THESE INSTRUMENTS ARE USED FOR TECHNICAL SPECIFICATION COMPLIANCE WHEN THE WATER TEMPERATURE IS LESS THAN 400°F.

COMBINING ALL OF THE ERRORS, IT WAS DETERMINED THAT; 1) THE ACTUAL FLOW MAY HAVE BEEN AS LOW AS 2026 GPM WHEN THE INDICATED VALUE WAS 3000 GPM, AND 2) THE RHR SYSTEM LOW-FLOW ALARM (INSTALLED AS PART OF A UNIT 2 LICENSING CONDITION) WOULD HAVE ACTIVATED AT APPROXIMATELY 675 GPM RATHER THAN 2000 GPM.

DURING AN INVESTIGATION OF THE ABOVE CONDITIONS, IT WAS DISCOVERED THAT DURING RHR SYSTEM OPERATION, WITHOUT A LOW FLOW ALARM, AN ALTERNATE FLOW PATH WAS SOMETIMES USED (SEE LER 315/85-040-00). THIS WAS DONE TO MINIMIZE PIPING VIBRATIONS THAT OCCURRED DURING RHR SYSTEM OPERATION. ONLY FLOW INDICATION IS PROVIDED WHEN THIS FLOW PATH IS UTILIZED.

EVALUATION OF DEGRADED FLOW CONDITION

EVALUATIONS HAVE BEEN DONE TO ASSESS THE EFFECT OF THE DEGRADED FLOW CONDITION ON

TEXT (If more space is required, use additional NRC Form 366A's) (17)

DECAY HEAT REMOVAL AND BORON STRATIFICATION. ALSO EVALUATED WERE THE CONSEQUENCES OF THE REDUCED LOW-FLOW ALARM SETPOINT. THESE EVALUATIONS HAVE SHOWN THAT AN ACTUAL FLOW OF 2000 GPM WOULD BE ADEQUATE TO REMOVE THE DECAY HEAT. THE ACTUAL FLOW RATE WITH THE INSTRUMENT ERROR WAS FOUND TO BE HIGHER THAN THE MINIMUM REQUIRED AS EVALUATED BY BOTH AEPSC AND WESTINGHOUSE (REFERENCE 18).

BORON STRATIFICATION WAS ASSESSED BY COMPARING THE TURBULENCE AND CORE CROSSFLOW THAT WOULD EXIST AT 2000 GPM TO WHAT EXISTS AT 3000 GPM. AT 3000 GPM (THE TECHNICAL SPECIFICATION LIMIT AND THE RHR PUMP DESIGN FLOW) BORON STRATIFICATION WOULD NOT OCCUR. THE EVALUATION SHOWED THAT THE REYNOLDS NUMBER IN THE DOWNCOMER OF THE REACTOR VESSEL IS IN THE TURBULENT REGION. TURBULENCE IN THE DOWNCOMER WOULD PROMOTE MIXING, THEREBY REDUCING ANY CONCENTRATION GRADIENTS THAT MAY HAVE EXISTED WHEN THE FLUID ENTERED THE DOWNCOMER. UPON ENTERING THE LOWER PLENUM OF THE REACTOR VESSEL, THE MOMENTUM OF THE FLUID COMBINED WITH THE EFFECTS OF A SUDDEN EXPANSION WOULD TEND TO ENTRAIN SURROUNDING FLUID, FURTHER REDUCING CONCENTRATION GRADIENTS. FINALLY, CROSSFLOW IN THE CORE WOULD PROMOTE ADDITIONAL MIXING. THE CROSSFLOW IS A FUNCTION OF THE REYNOLDS NUMBER TO THE 0.9 POWER (REFERENCES 2 AND 5). THE CROSSFLOW AT 2000 GPM WOULD BE 69 PERCENT OF THAT AT 3000 GPM. THUS, A SIGNIFICANT AMOUNT OF CROSSFLOW WOULD EXIST. (SEE ANALYSIS SECTION.)

ONCE THE FLOW EXITS THE CORE, THE RHR PIPING TURBULENCE WOULD BE VERY HIGH, AND CONSIDERABLE MIXING WOULD OCCUR, ESPECIALLY AS THE FLUID FLOWS THROUGH THE PUMP.

BASED ON THE FOREGOING MIXING EVALUATION, IT IS CONCLUDED THAT BORON STRATIFICATION IS NOT A CONCERN WITH 2000 GPM RHR FLOW.

BASED ON THE EVALUATION OF THE METER ERRORS, IT IS ESTIMATED THAT THE LOW-FLOW ALARM WOULD HAVE ACTIVATED AT 675 GPM RATHER THAN THE 2000 GPM FOR WHICH IT WAS SET. THE LOW-FLOW ALARM WAS SET SPECIFICALLY TO WARN THE OPERATOR THAT THE SHUTOFF OR ISOLATION VALVES HAD BEEN INADVERTENTLY CLOSED, AN ACTION WHICH WOULD REDUCE THE FLOW TO ZERO. THE OPERATOR WOULD THEN TAKE ACTION TO MITIGATE THE CONSEQUENCES OF THE EVENT. THE ACTUATION OF THE ALARM AT 675 GPM WOULD STILL SERVE THIS PURPOSE.

WHEN USING THE ALTERNATE FLOW PATH, IN WHICH NO RHR SYSTEM LOW-FLOW ALARM IS AVAILABLE, THE FLOW INDICATORS AND THE MOTOR AMPERAGE METERS WOULD ALERT THE OPERATOR TO A LOW-FLOW CONDITION. THUS, THE OPERATOR HAS THE CAPABILITY TO DETECT A LOW-FLOW CONDITION.

THE MOST CRITICAL TIME FOR A LOSS OF RHR FLOW TO OCCUR IS WHEN THE REACTOR IS DRAINED TO HALF-LOOP WHEN IN MODE 5 (COLD SHUTDOWN). UNDER THESE CONDITIONS, ONLY ONE RHR PUMP IS IN OPERATION (REFERENCE 3). THUS ONLY ONE PUMP COULD BE DAMAGED BY THE INADVERTENT CLOSURE OF THE PUMP SUCTION VALVES, AND THE OTHER WOULD BE AVAILABLE FOR SERVICE. (THIS IS CONSIDERED TO BE UNLIKELY BECAUSE POWER IS REMOVED FROM THE VALVES WHEN THEY ARE OPEN.)

THE LOSS OF RHR FLOW HAS BEEN EVALUATED IN THE FSAR (REFERENCE 15), AND THE TECHNICAL SPECIFICATIONS ALLOW THE REMOVAL OF RHR FLOW FOR UP TO ONE HOUR (REFERENCE 16). ON THIS BASIS, IT IS JUDGED THAT ADEQUATE TIME EXISTS FOR THE OPERATOR TO DETECT A LOW-FLOW CONDITION AND TAKE ACTION TO RESTORE COOLING WITHOUT THE ALARM.

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KEY ASSUMPTIONS

CALCULATIONS WERE PERFORMED TO DETERMINE THE MINIMUM RHR FLOW WHICH WOULD BE REQUIRED TO REMOVE DECAY HEAT. THE ASSUMPTIONS USED IN THE ANALYSIS WERE THAT THE LAKE WATER TEMPERATURE IS 85°F AND THE MAXIMUM TEMPERATURE OF THE REACTOR COOLANT WATER IS 200°F. THE LAKE SERVES AS THE ULTIMATE HEAT SINK FOR THE DECAY HEAT GENERATED IN THE CORE. THIS TEMPERATURE ULTIMATELY DETERMINES THE COOLANT FLOW TO THE REACTOR CORE REQUIRED. THE MAXIMUM REACTOR COOLANT TEMPERATURE IS SET BY THE TECHNICAL SPECIFICATION LIMIT OF 200°F IN MODE 5.

IT WAS ALSO ASSUMED THAT THE PRODUCT OF OVERALL HEAT TRANSFER COEFFICIENT AND SURFACE AREA (UA) WAS CONSTANT AND EQUAL TO THE DESIGN VALUE IN THE COMPONENT COOLING WATER HEAT EXCHANGER. A RATIO WAS COMPUTED FOR UA AS A FUNCTION OF REDUCED FLOW FOR THE RHR HEAT EXCHANGER. FLOW RATES OTHER THAN RHR FLOW RATE (SUCH AS COMPONENT COOLING WATER AND ESSENTIAL SERVICE WATER LOOPS) WERE ALSO ASSUMED TO REMAIN CONSTANT AND EQUAL TO DESIGN VALUES. CONSTANT PRESSURE SPECIFIC HEAT WAS TAKEN AS 1.0 BTU/LBM/°F FOR ALL FLOW STREAMS.

DECAY HEAT WAS CALCULATED USING REFERENCE 1 METHODOLOGY.

THE MINIMUM FLOW RATE WAS ALSO CALCULATED FOR AN ADDITIONAL CASE WITH 20 PERCENT MARGIN ON DECAY HEAT AND A LAKE WATER TEMPERATURE OF 95°F.

ANALYSIS

THIS SECTION PROVIDES DETAILS OF CALCULATIONS TO DETERMINE THE MINIMUM FLOW RATE REQUIRED TO REMOVE THE DECAY HEAT FROM THE REACTOR. THE RHR SYSTEM WAS MODELED USING THE FLOW DIAGRAM SHOWN IN FIGURE 1. THE PROBLEM INVOLVES SIX EQUATIONS AND SIX UNKNOWNS (THE TEMPERATURE OF EACH STREAM). THE BASIC EQUATIONS TO BE SOLVED ARE:

(1) $Q_i = M_i C_p \Delta T_i$

(2) $Q_i = U_i A_i \Delta T_{LM,\, i}$

EQUATION (1) DESCRIBES THE SENSIBLE HEAT GAIN OR LOSS IN THE COOLANT. EQUATION (2) DESCRIBES THE HEAT TRANSFER BETWEEN THE FLUIDS FLOWING ON THE SHELL SIDE AND THE TUBE SIDE OF THE HEAT EXCHANGER. THE LOG MEAN TEMPERATURE DIFFERENCE, ΔT_{LM} IN EQUATION 2, COMPENSATES FOR THE FACT THAT THE TEMPERATURE DIFFERENCE BETWEEN THE HOT AND COLD FLUID MAY CHANGE AS BOTH FLUIDS TRAVERSE THE HEAT EXCHANGER.

THE PRODUCT OF THE OVERALL HEAT TRANSFER COEFFICIENT, U, AND THE HEAT EXCHANGER SURFACE AREA, A, WAS DETERMINED FROM THE DESIGN CONDITION GIVEN IN TABLE 1. THIS WAS ACCOMPLISHED BY REARRANGING EQUATION 2 TO GIVE

(3) $UA = Q_j / \Delta T_{LM}$

THE CALCULATED VALUES OF UA ARE GIVEN IN TABLE 1.

THE DECAY HEAT USED IN THE EQUATION WAS DETERMINED BY

(4) $P/P_o\ (t_o,\ t_s) = P/P_o\ (\infty,\ t_s) - P/P_o\ (\infty,\ t_o + t_s)$

WHERE

P/P_o = POWER TO FULL-POWER RATIO
t_o = EFFECTIVE FULL POWER SECONDS AT 3411 MW
t_s = NUMBER OF SECONDS SINCE SHUTDOWN

(5) $P/P_o\ (\infty,\ t_s) = A\ t_s^{-a}$

WHERE A AND a ARE VALUES OBTAINED FROM REFERENCE 1.

BASED ON 1202 (REFERENCE 6) EFFECTIVE FULL-POWER-DAYS, THE DECAY HEATS WERE CALCULATED FOR DECAY TIMES OF 2.5 TO 6.0 DAYS. THESE RESULTS ARE GIVEN IN TABLE 2.

THE COMPONENT COOLING WATER HAS HEAT LOADS OTHER THAN THE DECAY HEAT FROM THE CORE. THE TOTAL AMOUNT OF THESE HEAT LOADS WAS OBTAINED FROM THE DESIGN VALUES AND WAS FOUND TO BE 34.9 (10^6) BTU/HR. FOR THE CALCULATION OF THE MINIMUM LOW FLOW THE DECAY HEAT AT 2.5 DAYS, 40.4 (10^6) BTU/HR WAS USED. THIS MADE THE TOTAL HEAT LOAD 75.3 (10^6) BTU/HR.

MASS FLOWS IN THE SYSTEM, (OTHER THAN RHR FLOW WHICH WILL BE CALCULATED), WERE OBTAINED FROM THE DESIGN VALUES. THE COMPONENT COOLING WATER FLOW WHICH IS DIVERTED TO THE AUXILIARIES IS SUMMARIZED (TABLE 3). THE MASS FLOWS USED IN THE CALCULATION ARE SUMMARIZED IN TABLE 4.

THE MINIMUM MASS FLOW RATE REQUIRED TO REMOVE DECAY HEAT AFTER 2.5 DAYS WITH A LAKE TEMPERATURE OF 85°F WAS DETERMINED BY ITERATION TO BE 1000 GPM.

TO ACCOUNT FOR UNCERTAINTIES IN THE DECAY HEAT VALUE, A MARGIN OF 20 PERCENT WAS ADDED WITH A LAKE TEMPERATURE OF 95°F AND THE CALCULATION REPEATED. WHEN THIS WAS DONE, THE MINIMUM REQUIRED FLOW WAS DETERMINED TO BE 1450 GPM.

THE QUESTION OF BORON STRATIFICATION WAS ADDRESSED BY COMPARING THE REYNOLDS NUMBERS AND MIXING PARAMETERS FOR 2000 GPM AND 3000 GPM RHR FLOW. THERE ARE SEVERAL PLACES IN THE PIPING WHERE MIXING COULD OCCUR. THESE INCLUDE THE REACTOR COOLANT SYSTEM PIPING, THE REACTOR VESSEL DOWNCOMER, THE REACTOR VESSEL UPPER AND LOWER PLENUM, THE CORE REGION, AND THE RHR SYSTEM PIPING. THE REYNOLDS NUMBER FOR THESE REGIONS IS SHOWN IN TABLE 5. EXCEPT FOR THE PLENUMS (FOR WHICH NO CALCULATION WAS MADE), AND THE CORE REGION, THE REYNOLDS NUMBER EXCEEDS 4000. BECAUSE OF THIS, IT IS CONCLUDED THAT FLOW WOULD BE TURBULENT AND THAT ADEQUATE MIXING WOULD OCCUR.

IN THE CORE REGION, FLOW IS LAMINAR. HOWEVER, THERE IS MIXING DUE TO CROSSFLOW WITHIN THE CORE REGION. A MIXING PARAMETER

TEXT (If more space is required, use additional NRC Form 366A's) (17)

$$(6)\quad \beta = \frac{G}{\bar{G}} = K\,R_e^{-0.1}$$

EXISTS WHICH RATIOS CROSSFLOW IN THE CORE TO THE AVERAGE CORE FLOW (REFERENCE 2). SINCE THE REYNOLDS NUMBER IS DIRECTLY PROPORTIONAL TO THE FLOW IN THE SYSTEM, THE EQUATION CAN BE MODIFIED TO GIVE

$$(7)\quad G = K\,(\bar{G})^{0.9}$$

FROM THIS, THE CROSSFLOW AT TWO DIFFERENT FLOW RATES CAN BE COMPARED.

$$(8)\quad \frac{G_2}{G_1} = \left[\frac{\bar{G}_2}{\bar{G}_1}\right]^{0.9}$$

THE TECHNICAL SPECIFICATIONS REQUIRE A MINIMUM OF 3000 GPM TO PREVENT STRATIFICATION, AND THE ACTUAL FLOW THROUGH THE SYSTEM MAY HAVE BEEN AS LOW AS 2000 GPM. USING EQUATION 8, IT CAN BE SEEN THAT THE CROSSFLOW AT 2000 GPM RHR FLOW WOULD BE APPROXIMATELY 69 PERCENT OF THAT AT 3000 GPM - $(2000/3000)^{0.9}$. THIS CROSSFLOW, TOGETHER WITH THE MIXING THAT WOULD OCCUR IN THE PIPING AND THE DOWNCOMER, IS JUDGED TO BE SUFFICIENT TO PREVENT SIGNIFICANT BORON CONCENTRATION.

IT IS CONCLUDED THAT THE LOW RHR FLOW DOES NOT CONSTITUTE AN UNREVIEWED SAFETY QUESTION AS DEFINED IN 10 CFR 50.59, NOR DID THEY CREATE A SUBSTANTIAL HAZARD TO THE HEALTH AND SAFETY OF THE PUBLIC.

TABLE 1

HEAT EXCHANGER DESIGN CONDITIONS

RHR HEAT EXCHANGER	
DESIGN HEAT LOAD, BTU/HR	41.1×10^6
SHELL SIDE INLET TEMPERATURE, °F	95.
TUBE SIDE INLET TEMPERATURE, °F	140.
SHELL SIDE OUTLET TEMPERATURE, °F	111.6
TUBE SIDE OUTLET TEMPERATURE	112.3
CALCULATED UA, BTU/HR °F	1.836×10^6
CCW HEAT EXCHANGER	
DESIGN HEAT LOAD, BTU/HR	76×10^6
SHELL SIDE INLET TEMPERATURE, °F	114.
TUBE SIDE INLET TEMPERATURE, °F	76.
SHELL SIDE OUTLET TEMPERATURE, °F	95.
TUBE SIDE OUTLET TEMPERATURE, °F	92.
CALCULATED UA, BTU/HR °F	3.71×10^6

TABLE 2

DECAY HEAT AS A FUNCTION OF TIME

TIME AFTER SHUTDOWN, DAYS	DECAY HEAT, 10^6 BTU/HR
2.5	40.4
3.0	38.1
3.5	36.2
4.0	34.6
4.5	33.3
5.0	32.1
5.5	31.1
6.0	30.2

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TABLE 3

AUXILIARY COOLING WATER FLOWS

COMPONENT	FLOW, GPM
REACTOR COOLANT PUMP	560
SEALWATER HEAT EXCHANGER	38
LETDOWN HEAT EXCHANGER	300
SPENT FUEL HEAT EXCHANGER	1500
RHR PUMP	10
SI PUMP	40
SPRAY PUMP	20
CHARGING PUMP	90
PENETRATIONS	300
GAS COMPRESSOR	13
REACTOR SUPPORT	40
TOTAL	2911

(REFERENCE 14)

TABLE 4

MASS FLOWS USED IN ANALYSIS

FLOW STREAM (REFER TO FIGURE 1)	MASS FLOW, 10^6 LB/HR
M_1	TO BE CALCULATED
M_2	2.56
M_3	4.67
M_4	4.0

(REFERENCES 12, 13, 14)

TABLE 5

REYNOLDS NUMBERS

LOCATION	REYNOLDS NUMBER 3000 GPM	2000 GPM
REACTOR INLET	631,500.	421,000.
REACTOR OUTLET	598,836.	399,224.
REACTOR DOWNCOMER	27,000.	18,000.
REACTOR CORE	1,020.	850.
RHR PIPING	2,880.000.	1,920,000.

(REFERENCES 7, 8, 9, 10, 11, 17)

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REFERENCES

(1) NUREG/CR-2507 "BACKGROUND AND DERIVATION OF ANS 5.4 STANDARD FISSION PRODUCT RELEASE MODEL," JANUARY, 1982.

(2) XN-NF-81-73, "TURBULENT MIXING IN ROD BUNDLES," OCTOBER, 1982.

(3) PMP OHP 4021.017.002, "PLACING THE RHR IN SERVICE."

(4) DCC HP 106, "RHR SYSTEM DESCRIPTION".

(5) McCABE AND SMITH, "UNIT OPERATIONS OF CHEMICAL ENGINEERING," 1956, P. 53.

(6) NUCLEAR FUEL RELOAD SPECIFICATION FOR D.C. COOK PLANT, SPEC. NO. DCCNE-104-QCN, OCTOBER 1984, REVISION 2.

(7) FSAR, TABLE 3.2.1-1.

(8) FSAR, TABLE 3.4-1.

(9) FSAR, TABLE 4.1-3.

(10) DWG 1-5143, FLOW DIAGRAM EMERGENCY CORE COOLING (RHR), REVISION 30, 5-7-85.

(11) DWG 2-5143, FLOW DIAGRAM EMERGENCY CORE COOLING (RHR), REVISION 29, 10-25-84.

(12) FSAR, TABLE 9.8-5.

(13) FSAR, TABLE 9.5-3.

(14) FSAR, TABLE 9.5-2.

(15) FSAR, SECTION 9.3.

(16) TECHNICAL SPECIFICATION 3.4.1.3.

(17) DWG 11-5128 FLOW DIAGRAM REACTOR COOLANT, REVISION 16, 3-21-85.

(18) WESTINGHOUSE PRELIMINARY EVALUATION, RHR REDUCED FLOW IN MODE 5, AUGUST, 1985.

NOMENCLATURE

Qi	=	HEAT FLOW	BTU/HR
M_i	=	MASS FLOW RATE FOR STREAM i	LB/HR
C_p	=	HEAT CAPACITY	BTU/LB °F
T_i	=	TEMPERATURE DIFFERENCE FOR STREAM i	°F
U_i	=	OVERALL HEAT TRANSFER	BTU/HR FT^2 °F
A_i	=	HEAT TRANSFER AREA FOR HEAT EXCHANGER i	FT^2
$T_{LM, i}$	=	LOG MEAN TEMPERATURE DIFFERENCE FOR HEAT EXCHANGER i	°F
p	=	ACTUAL REACTOR POWER	MW
P	=	RATED REACTOR POWER	MW
t	=	EFFECTIVE FULL-POWER OPERATING TIME	DAYS
t_s	=	TIME SINCE SHUTDOWN	DAYS
G	=	CROSSFLOW	LB/FT^2 SEC
G	=	AVERAGE COOLANT FLOW	LB/FT^2 SEC
β	=	MIXING PARAMETER	DIMENSIONLESS

NRC Form 366A
(9-83)

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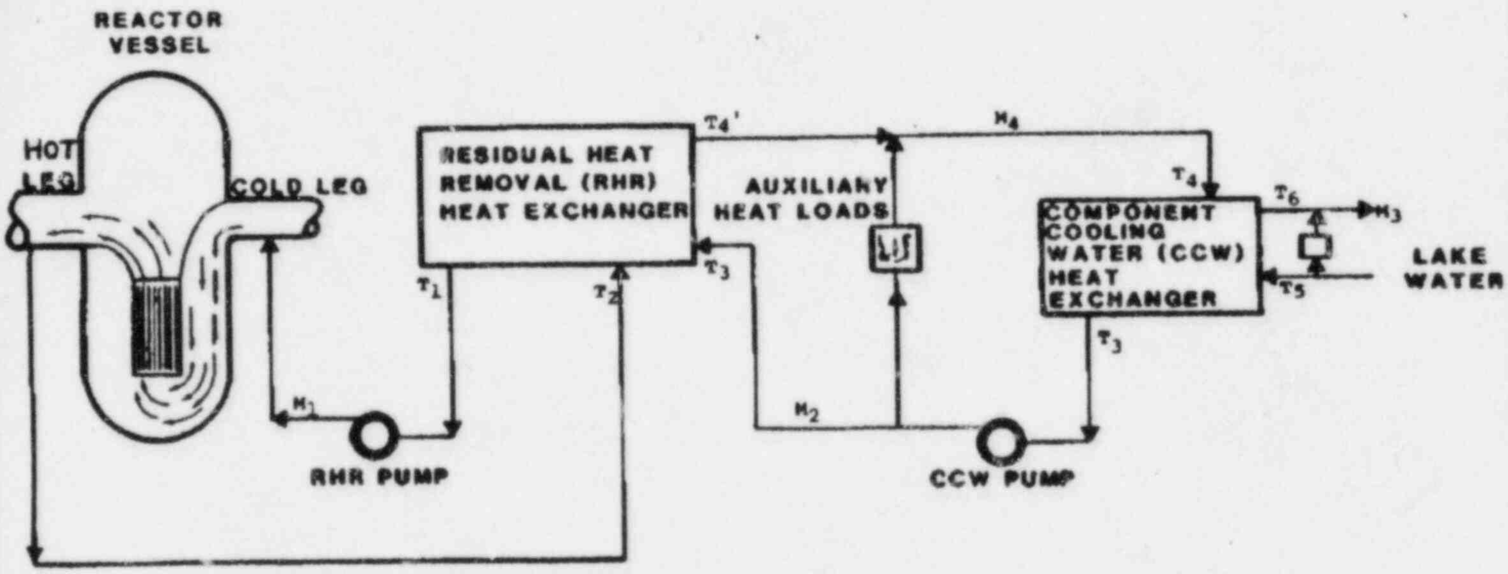
FIGURE 1. RHR LOOP CONFIGURATION MODEL



American Electric Power System
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September 9, 1985

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Docket No. 50-315

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In accordance with the criteria established by 10CFR50.73 entitled Licensee Event Reporting System, the following report/s are being submitted:

RO 85-031-01

Sincerely,

B.A. Svensson
for W.G. Smith, Jr.
Plant Manager

/cbm

Attachment

cc: John E. Dolan
J.G. Keppler, RO:III
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R.F. Kroeger
H.B. Brugger
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NRC Resident Inspector
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