



May 17, 1990 3F0590-06

U.S. Nuclear Regulatory Commission Attn: Document Control Desk Washington, D.C. 20555

> 9005240103 90057702 PDR ADOCK 05000

FDC

Subject: Crystal River Unit 3 Docket No. 50-302 Operating License No. DPR-72 Reactor Building Flooding

Reference: Licensee Event Report No. 90-005, dated April 29, 1990

Dear Sir:

On March 29, 1990, Florida Power Corporation (FPC) determined that Crystal River Unit 3 (CR-3) was operating outside the plant design basis due to procedures which were based upon a non-conservative Reactor Building (RB) maximum accident flood level calculation. This problem is described in the reference LER (copy enclosed). The purpose of this letter is to describe the resolution which FPC will pursue.

The Borated Water Storage Tank (BWST) is the primary source of ECCS water to mitigate design basis accidents. During the initial phase of LOCA mitigation, the BWST provides the source of water for the High Pressure Injection (HPI), Low Pressure Injection (LPI) and Building Spray (BS) pumps to inject water into the reactor vessel and spray into the RB atmosphere. The water inventory from the reactor coolant system, the BWST, and the other tanks (Core Flood Tank, Makeup Tank, and Sodium Hydroxide Tank) fall to the RB floor, eventually reaching the RB sump located in the 95 ft floor elevation (plant datum). Flooding of this elevation will occur. In accordance with the current emergency operating procedures, when the BWST level reaches 2.5 ft, the operator manually transfers the LPI and BS pump suctions from the BWST to the RB sump, initiating the recirculation phase. This action would result in a RB flood

POST OFFICE BOX 219 . CRYSTAL RIVER, FLORIDA 32629-0219 . (904) 563-2943 A Florida Progress Company

May 17, 1990 3F0590-06 Page 2

level of approximately 102 ft elevation with maximum Technical Specification volumes in the various tanks at the beginning of the accident. A number of instruments necessary to mitigate the design basis accidents are installed on the walls below the 102 ft elevation, but above the 99.85 ft elevation (the maximum flood level described in FSAR Section 6.2.2.1). Thus, a design basis accident could cause these instruments to be submerged. It is, therefore, necessary to initiate operator action to limit the flood level to the FSAR value of 99.85 ft elevation.

To limit the flood level, there are three considerations which must be addressed. These are:

- Water volumes and boron concentrations for core cooling and shutdown requirements must be satisfied,
- b. Minimum required NPSH of the LPI and BS pumps must be satisfied before switchover to sump, and
- c. Materials inside the RB must be compatible with the pH of the RB spray or sump fluid.

The licensing bases LOCA models consider the first item and limit the minimum volume of water and boron concentration in the BWST. The required levels for the LPI pumps and the BS pumps NPSH are 95.6 ft and 97.0 ft, respectively. Maintenance of the expected pH in the range of 7.2 to 11.0 by establishing a proper balance of sodium hydroxide and borated water will assure material compatibility. A limited duration excursion above a pH of 11.0 is verified to be acceptable in our EQ program documentation.

The approach which best satisfies these considerations and resolves the RB flooding issue is to limit the volume of water contributed by the BWST. This will be accomplished by a procedural change in operator action. The operator will ensure the RB flood level will not exceed the 99.85 ft elevation. Following & LOCA, the operator will begin manual transfer of the ECCS pump suctions from the BWST to the RB sump when he receives an alarm indicating that the RB flood level has reached approximately 97.6 ft elevation (this level allowance for instrumentation error). includes an The corresponding actual level will satisfy the ECCS pump NPSH, core cooling, shutdown, and pH requirements. The RB flood level will be less than 99.85 ft with the switchover from the BWST to the RB sump completed in 10 minutes even under worst case large break LOCA flow rates. FPC has confirmed that this switchover can be accomplished in this amount of time.

Prior to CR-3 restart from this refueling outage, FPC will perform the following actions:

May 17, 1990 3F0590-06 Page 3

- 1. A modification will be installed to add an alarm in the Main Control Room to indicate when the RB flood level has reached the point that operator action to initiate switchover from the BWST to the sump is to begin.
- Operational procedures will be revised and issued. Operator training on the use of the RB flood level as the new switchover parameter will be completed.

If you have any questions, please contact Mr. Rolf C. Wide Director, Nuclear Site Support at (904)563-4329.

Sincerely,

Bearth

P. M. Beard, Jr Senior Vice President Nuclear Operations

PMB/JWT

Enclosure

xc: Legional Administrator, Region II Senior Resident Inspector

Packet Hill CRYSTAL RIVER UNIT 3 Doct Market Billion 1 (2010) [2010] [2010	MAC PO	186 MA		LICI	ENSEE EV	ENT	REP	ORT (LER)	GULATOR	Y COMMIE	ESTIMA INFORM COMMET AND RE REGULA THE FA	APPROVI EI TED BURDEN PE ATION COLLECT VIS REGARDING PORTS MANAGE ITORY COMMISSI PERWORK REDU AGEMENT AND	ID OM	NO 3150-010 S 4/30/02 SPONSE TO CO REQUEST 501 EN ESTIMATE BRANCH (P 5) (ASHINGTON V PROJECT (3) T WASHINGTO	M DMPLY WT D HR. FOI TO THE RE ID: US NU DC 20555 J ISO 01041 DN DC 2056	H THIS RWARD ICORDS ICORDS ICLEAR AND TO OFFICE 03
This if Constrained in the second state of the second state second state of the	ACILIT	-	0	65.V.6									DOCKET NUMBE	A (2)		PAS	10
Call Coll Coll Coll Coll Coll Coll Coll	-	- Ca	louls	CRYS	F PR FL	RUN	IT J	1 Sho	we to	vel F	veed	that of	0 15 10 10	010	131012	1 OF	1010
<th< td=""><td></td><td>to</td><td>Desi</td><td>gn Er</td><td>ror Caus</td><td>ing</td><td>Oper</td><td>ation</td><td>Outs</td><td>ide D</td><td>esign</td><td>Basis</td><td>. sale on</td><td></td><td>own Equ</td><td>1 pluen</td><td>L Due</td></th<>		to	Desi	gn Er	ror Caus	ing	Oper	ation	Outs	ide D	esign	Basis	. sale on		own Equ	1 pluen	L Due
UNATE CALL VIAN	EVI	INT DATE	(6)	T	SEQUENTIA	(6)	NE VIBION	RE	ORT DAT	1 (7)		OTHER	FACILITIES INV	OLVE	D (8)		******
0 3 29 9 0 9 0 - 0 0 5 - 0 0 4 3 0 9 0 N/A 0 5 0 1 0 1 0 1 1 - 1 - 1 - 1 - 1 - 1 - 1 -	MONTH	DAY	YEAR	YEAR	NUMBER	++		MONTH	DAY	YEAR		N/A		0	151010	101	
Order the action of the statement of the stateme	0 3	2 9	910	9 0 -	- 0 0 5	-	00	94	3 0	90		N/A		0	151010	101	
All Description Provide setup:	001	RATINO	4		AT IS SUDMITT	D PUR	BUANT	-	OUIREM		CFR § /0	check one or more	of the following)	(11)			
NAME D 0 0 D meaning Description Description <thdescription< th=""> Description <thdes< td=""><td></td><td></td><td></td><td>20.40</td><td>2(6)</td><td></td><td>-</td><td>20.4061</td><td>•1</td><td></td><td>-</td><td>60.73(a)(2)(iv)</td><td></td><td>H</td><td>73.71(6)</td><td></td><td></td></thdes<></thdescription<>				20.40	2(6)		-	20.4061	•1		-	60.73(a)(2)(iv)		H	73.71(6)		
NAME Description Description <thdescription< th=""> <t< td=""><td>LEVE</td><td>0</td><td>010</td><td>20.40</td><td>6 (a)(1)(N)</td><td></td><td></td><td>60.38 (e</td><td>1(2)</td><td></td><td></td><td>50.73(a)(2)(+#)</td><td></td><td>E</td><td>OTHER IS</td><td></td><td>atract</td></t<></thdescription<>	LEVE	0	010	20.40	6 (a)(1)(N)			60.38 (e	1(2)			50.73(a)(2)(+#)		E	OTHER IS		atract
Participation X B0 3001000 B0 3001000 NAME L. W. MOFFATT, NUCLEAR SAFETY SUPERVISOR TELEMONE NUMBER COMPLETE ONE LINE FOR EACH COMPONENT PAILURE DECRETOR IN THE REPORT (IS) TELEMONE NUMBER CAUSE SYSTEM COMPONENT MANUAR A I I I I N/A I I I I I BUPGLEMENTAL AREPORT EXPECTED (IN) SUBJECTED SUBJECTED SUBJECTED SUBJECT TOWING THE INFORMATION INFORMAT				20.40	6(+)(1)(W)		F	\$0.73ta	(2)())			60.73(a)(2)(vil)	(▲)	Γ	366A)		
AMME L. W. MOFFATT, NUCLEAR SAFETY SUPERVISOR TELEPHONE NUMBER COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DECRETO IN THE REPORT IS COMPONENT YELEPHONE NUMBER CAUSE SYSTEM COMPONENT MANUAC TELEPHONE NUMBER CAUSE SYSTEM COMPONENT MANUAC TELEPHONE NUMBER A I I I I I N/A I I I I I I N/A I I I I I I I N/A I I I I I I I I N/A I <				20.40	B(a)(1)(iv)		×	60.736	(2)(H)		-	00.73(a)(2)(a)	(•)				
NAME TelePhone Humster L. W. MOFFATT, NUCLEAR SAFETY SUPERVISOR Image: Second Seco								ICENSEE	CONTACT								
L. W. MOPPATT, NUCLEAR SAFETY SUPERVISOR 9 1 014 7 1 915 1- 16 1 418 CONFICTEONE LINE FOR LACH COMPONENT FAILURE DESCRIPTION THE REPORT FAIL CAUSE SYSTEM COMPONENT WATURE TO ATALLE CAUSE SYSTEM COMPONENT WATURE TO ATALLE A 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1	NAME												ABEA (00	37	LEPHONE NUN	486.	
Constitutions of the Net Positive Superating of the the standard for the s		L.	W. M	OFFATT	r, NUCLEA	AR S	AFET	Y SUP	ERVIS	OR			0.00			1614	1014
CAUSE SYSTEM COMPONENT MANUAC TUREA PERMITAL TO MARDS CAUSE SYSTEM COMPONENT MANUAC TUREA PERMITAL TO MARDS A I <td></td> <td></td> <td></td> <td></td> <td>COMPLETE</td> <td></td> <td></td> <td>EACH CO</td> <td>MPONEN</td> <td>FAILURE</td> <td>DESCRIBE</td> <td></td> <td>AT (13)</td> <td>• 1/</td> <td>1 71 2 1-</td> <td>1014</td> <td>1010</td>					COMPLETE			EACH CO	MPONEN	FAILURE	DESCRIBE		AT (13)	• 1/	1 71 2 1-	1014	1010
A Image: A indication of the Net Positive Successful in the indication of the Net Positive Successful in the indications and calculation of the Net Positive Successful in 1972 did not identify this problem because the design engineers assumed the original calculation. FPC is evaluating possible solutions to resolve this design basis is successful in the indication is provided after 1972 did not identify this problem because the design engineers assumed the original calculation. FPC is evaluating possible solutions to resolve this design basis is successful in the indication is provided after 1972 did not identify this problem because the design engineers assumed the original calculation was correct or were unaware of the original calculation. FPC is evaluating possible solutions to resolve this design basis is successful in the refueling outage and is continuing to evaluate corrective actions to prevent recurrence.	CAUSE	SYSTEM	COMPO	INENT	MANUFAC TURER	10	ATABLE			CAUSE	SYSTEM	COMPONENT	MANUFAC		TO NPRDS		
N/A Image: Sum Content of the point state of the point poin	A			4	111	1		ļ		·	<u> </u>		1	4			
EXPECTED SUBMISSION DATE: NO MONTH DATE TO SUBMISSION DATE: ABSTRACT (LIMM REFECTED SUBMISSION DATE: NO NO 111159 ABSTRACT (LIMM REFECTED SUBMISSION DATE: NO NO 111159 ABSTRACT (LIMM REFECTED SUBMISSION DATE: NO 111159 ABSTRACT (LIMM REFECTED SUBMISSION DATE: NO 111159 111159 ABSTRACT (LIMM REFECTED SUBMISSION DATE: NO 111150 111150 ABSTRACT (LIMM REFECTED SUBMISSION DATE: NO 111150 111150 ABSTRACT (LIMM REFECTED SUBMISSION DATE: NO 1111150 111150 ABSTRACT (LIMM REFECTED SUBMISSION DATE: NO 1111150 111150 ABSTRACT (LIMM REFECTED SUBMISSION DATE: NO 11111150 1111150 1111500	N/A	1	1.1	1	111						1.	111	111	,			
X YES III , an company EXPECTED SUBMISSION DATE: ABETRACT (Limit in 1400 uncent is separating that approximately 1400, Florida Power (FPC) determined that Crystal River Unit 3 (CR-3) was operating outside the plant design basis due to a non-conservative Reactor Building (RB) maximum accident flood level calculation. When re-calculated using conservative assumptions, the maximum RB flood level exceeds the level necessary to prevent submergence of safe shutdown instrumentation and equipment. At the time of this determination, CR- 3 was in MODE 5 (COLD SHUTDOWN) preparing for a refueling outage. This non- conformance was caused by an oversight in 1972 by the design engineer when a calculation of the Net Positive Suction Head for post-accident recirculation cooling was incorrectly used for the maximum flood level. Modifications and calculations performed after 1972 did not identify this problem because the design engineers assumed the original calculation was correct or were unaware of the original calculation. FPC is evaluating possible solutions to resolve this design basis issue prior to restart from the refueling outage and is continuing to evaluate corrective actions to prevent recurrence.					SUPPLEM	ENTAL	-	EXPECT	0 114		A		ExPE	TED	MONT	H DAY	YEAR
AMATTANCY (Limit to 1400 meru): a consistency of the set of the se		111	mains E)		INISSION DAT			+	7				DATE		1.1	1.5	9.0
	ABETRAC	On Cry to cal RB shu 3 w con cal coo cal des of thi con	March stal a nor culat flood tdown as in forma culat ling culat ign e the c s des tinui	A 29, River n-cons tion. 1 leven inst MODE ance w tion o was 1 tions engine origin tign b ng to	1990 at Unit 3 ervative When re 1 exceed rumentat 5 (COLD as cause f the Ne ncorrect performe ers assu al calcu asis iss evaluat	apprice appric	roxin -3) in actor lcula ine lo and UTDO y an osit used fter the ion. orion	matel, was o r Bui ated evel equi ive S for 1972 orig FPC r to ctive	y 140 perat lding using neces pment repar sight uctio the m did inal is e resta acti	0, Flo ing of (RB) conse sary f . At f ing fo in 19 n Head aximum not io calcul valuat ons to	brida utside maxim ervati to pre- the ti- pra floc dentif lation ting p bm the p pre-	Power (F e the pla num accid ve assum event sub me of the refueling the des post-acc od level. fy this p was cor ossible refueli e refueli	PC) dete int desig lent floo options, mergence is deter outage. ign engi ident re Modifi problem b rect or solution ing outag irrence.	rmin d l the of min T cat eca wer t e a	ned tha asis du evel maximu safe ation, his nor culation ions ar use the culatic ions ar use the cunawa co resol	cR- a in id ire ve	

LICENSEE EVENT REP TEXT CONTINUAT	ORT (LER)		APPROVED 34 Expire TED SURDEN PER RE ATION COLLECTION NTS REGARDING SURD INONTS MANAGEMENT ATORY COMMISSION Y PERWORK REDUCTION VACEMENT AND BUDG	SANCH IN ST	0 COM U 500 HAS 500 HAS 500 HAS 500 U 500 U 0 DC 70 1 J15001 WGTON DC		CORDS COLGAR IND TO DS I ICE
	DOCKET NUMBER (2)				•		31
		-148	SEQUENTIAL			Π	
CRYSTAL RIVER UNIT 3	0 5 0 0 0 3 0 2	9 0	_ 0 0 1 5 _	010	012	OF	0 16
TUT IN more spece a mounted up addressed NRC form MA (1)(1)				1-1-		101	

EVENT DESCRIPTION

On January 9, 1990, during an engineering review of a calculation performed for NRC Bulletin 79-01B, non-conservative assumptions were identified in the calculation of the Reactor Building (RB) [NH] flood level. This calculation review was performed as a part of the engineering configuration management verification and upgrade. The assumptions included using nominal or minimum tank volumes instead of maximum tank volumes permitted by Technical Specifications and assuming water from inside the primary shield wall does not reach the RB sump [NH,ACC]. Based on these concerns, the maximum RB flood level was re-calculated. The final results of the corrected calculations were received by Florida Power on March 28, 1990. Evaluation of the results on March 29, 1990, at approximately 1400, concluded the maximum level would exceed the level necessary to prevent submergence of constituted to be outside the plant design basis.

At the time of the verification that the plant was outside the design basis, March 29, 1990, CR-3 was in MODE 5 (COLD SHUTDOWN) in preparation for a refueling outage. No immediate actions were necessary.

At 1410 on March 29, 1990, a four-hour verbal report of this non-conformance was provided to the NRC Operations Center per 10CFR50.72(b)(2)(i) requirements. This written report is being provided per the requirements of 10CFR50.73(a)(2)(ii).

CAUSE

This non-conformance was caused by an oversight in 1972 by the architect design engineer performing the original RB flood level calculation. The original calculation used minimum or nominal Borated Water Storage Tank (BWST) [BP,TK], NaOH Tank [BE,TK], and Core Flood Tanks (CFT) [BP,TK] volumes and assumed that water within the primary shield area and other areas does not reach the RB sump area. These assumptions are appropriate for calculating the minimum water level available for Decay Heat Removal and Building Spray systems pump [BP,P][BE,P] Net Positive Suction Head. The resultant minimum water level of 99.85 ft. plant datum was incorrectly assumed to be the maximum level above which critical instruments and equipment must be located.

Since the original 1972 calculation, other calculations and modifications have occurred which could have identified this problem. In 1981, during evaluation of NRC Bulletin 79-01B, the maximum RB flood level was reviewed and recalculated. These calculations were based on the faulty assumptions and incorrectly calculated a new flood level. In 1987, the primary shield wall was modified when drain holes were drilled in the wall to allow water to drain to the RB sump area. The utility Design Engineer contacted the Nuclear Steam Supply System vendor, the Architect Engineering firm, and a sister utility to determine why drains

LICENSEE EVENT REI TEXT CONTINUA	PORT (LER)	EST UN	1100 MATIC MATIC ATOR ATOR ATOR	APRO		0 NC 315	0 00 04 0 00 04 000 46 0 100 46 0 100 0 0 100 0 0 100 0 0 100 0 0 100 0	- 01 	- 1418 CORDS	「「「「「「「」」」」」」」」」」」」」」」」」」」」」」」」」」」」」」
174 NA06 (1)	DOCKET NUMBER (2)		LER	NUMBER	(0)		,	AGE	2	-
•		*140	L.		-			TI		-
CRYSTAL RIVER UNIT 3	0 5 0 0 0 3 0 2	9 10	_	0 0 0	5 -	010	013	OF	01	6
The second a second of second sec news sea si (17)										1

were not already installed, but no reason could be provided. The utility design engineer did not determine any impact on the original flood calculation since the impact to the original, or later calculations and assumptions was not provided. In 1989, the flood level calculation was reviewed by a contract design engineer to determine the impact of the addition of equipment which would displace water and thus potentially raise the maximum water level. This calculation did not re-verify the previous level calculation, but simply evaluated the impact on the level. In conclusion, it appears these calculations and modifications did not identify the faulty assumptions because the design engineers assumed the original calculation was correct or were not provided the pertinent original design assumptions.

EVENT EVALUATION

...............

FSAR Section 6.2.2.1 states:

"In the event of a postulated LOCA [Loss of Coolant Accident], water will be pumped into the reactor building via the Reactor Building Spray System and Decay Heat Removal System as described in Sections 6.2 and 9.4, respectively. The reactor building will fill to an approximate elevation of 99.85 ft. prior to the initiation of the recirculation mode of the Emergency Core Cooling System (ECCS)."

This statement is incorrect. The essential safe shutdown instrumentation used during a LOCA is located approximately two to three inches above the 99.85 ft. elevation. The correct flood level elevation should be 101.7 ft.(101'8"), approximately 1.85 feet above the incorrect flood level. As a result, the attached list of essential safe shutdown equipment may be subjected to an environment for which they are not qualified to perform their safety function.

¥.

The corrected flood level elevation of 101'8" is based on maximum BWST, NaOH, and CFT tank volumes and assumes the water level in the primary shield wall will equalize with the level in the RB sump area following a LOCA in the cold leg reactor coolant pump [AB,P] suction. Additionally, the entire contents of the Reactor Coolant System (RCS) [AB], less the Reactor Vessel [AB,RPV] volume, are assumed to contribute to the final maximum water level. These new assumptions are conservative because few accidents result in totally draining tanks and major portions of the Reactor Coolant System.

Most of the instrumentation and equipment affected will perform their automatic safety function before the water level reaches the equipment. Automatic reinitiation of High Pressure Injection (HPI) [BQ] may not occur. Additionally, automatic actuation of Low Pressure Injection (LPI) [BP] may not occur. However, operators would be capable of manually initiating these safety systems. If the affected containment isolation valves [JM, ISV] have been opened, the associated

LICENSEE EVE	NT REPORT (LER)	51 00 0000000000000000000000000000000000	APRACUED ONG SC JISCOTA ETFICES 4 10 82 ESTIMATED BURDEN FER REPONSE TO COMPLY MTM. INFORMATION COLLECTION REDUEST SCO HAS FORM COMMENTS RECARDING BURDEN ESTIMATE TO THE RECO AND REPORTS MANAGEMENT BRANCH IP SID: US NUCL RECULATORY COMMISSION WASHINGTON DC 20055 AME THE PARENWORE REDUCTION PROJECT JISO CIDAL OF DF MANAGEMENT AND BUDGET WASHINGTON DC 20053							
ILITY NAME (1)	DOCKET NUMBER (2)		LEC	NUM	6 A (6)					
CRYSTAL RIVER UNIT 3		*140			1 41			1	T	
	0 5 0 0 0 3 0 2	9 10	-	0 10	15	-	00	04	05	0 16

containment penetration can still be automatically isolated with the isolation valve located outside the RB.

One impact of the new flood level is the information provided to the operator. Because many RCS pressure instruments [AB,PIT] are affected, the operator may not have reliable pressure and subcooling margin information.

CORRECTIVE ACTIONS

60

Č.

Florida Power is evaluating possible solutions to resolve this non-conformance prior to restart from the current refueling outage. These solutions include relocation of equipment necessary for accident mitigation, re-evaluation of the analytical methods which have established operator procedural steps, and reduction in tank volumes.

Florida Power is evaluating the modification and calculation controls for improvement to reduce the possibility of the types of errors which contributed to this non-conformance, in future modifications and calculations.

PREVIOUS SIMILAR EVENTS

This is the first report related to design error in the maximum RB flood elevation calculation. Two prior events were identified that also relate to RB equipment inoperability due to submergence. These previous events were concerned with locating equipment below the established RB flood elevation.

LICENSEE	EVENT REPOR	T (LER)		4 110 4 10 4 10 4 10	LUNDEN M BURDEN M DON COLLEC IN CARDING RTS MANAGE AV COMMISS RWORK RED IMENT AND		AL DUST	0 COMPLY 500 HAS 418 TO THI 0 DC 708 T 3150010 NGTON DC	NUCLE NUCLE SS AND NUCLE SS AND NO		
		DOCKET NUMBER (2)			-		*AGE 1				
CRYSTAL RIVE	R UNIT 3	0 ;5]0]0]0]3] 0] 2	910	-	0 0 1 5	5 -	-010	015	0 0		
	SAFE SHU	TDOWN EQUIPMENT AFFECT	ED								
TAG NO.	EQUIPMENT D	ESCRIPTION									
INSTRUMENTATION											
AH-536-TE [NH, TI]	Reactor Building temperature instrumentation, used for post accident monitoring										
RC-1-LT1 [AB,LT] RC-1-LT3	Pressurizer level transmitters. This instrumentation is used by the operator and by the Integrated Control System to control makeup flow and pressurizer level.										
RC-3A-PT3 [AB,PT] RC-3A-PT4 " RC-3B-PT3 "	Reactor Coo o Reactor I pressure and o Engineer ion and Low o Automatic of the Decay o Pressuri Valve for Ri o Main Co Subcooling N	lant System (RCS) pres Protection System for d variable low pressur ed Safeguards System f Pressure Injection au c Closure Interlock fo y Heat Removal System. Zer spray, heaters a CS pressure control. Introl Board indicati Margin.	sure high e Rea or Hi tomat r ove ad P on f	tr pr act igh tic erp ilc or	ransmit ressure for trip Press cactua pressure ot Oper RCS p	ter , 1 ps. ure tio e p rat	s used ow Injed n. orotect ed Re ssure	d by ct- tion lief and			
RC-132-PT [AB,PT]	Low range Ri indication safeguards a	CS pressure transmitte on the Main Control actuation.	rs us Board	sed	for pland for	res r e	sure	ered			
RC-158-PT [AB,PT] RC-159-PT	Wide range F Control Boar	RCS pressure transmitt rd and the Remote Shut	ers i down	Fa	ated on anel.	n t	he Ma	in			
RC-163A-LT1[AB,PT] RC-163B-LT1 RC-164A-LT1 RC-164B-LT1 RC-164B-LT1 RC-163A-TE1[AB,TT] RC-163B-TE1 RC-164A-TE1 RC-164B-TE1	Reactor Cool temperature	lant Inventory Trackin transmitters used for	g Sys post	ste - ac	em leve ccident	1 a in	ind dicat	ion.			
WD-303A-LT[NH,LT] WD-303B-LT " WD-304A-LT " WD-304B-LT "	Reactor Buil	lding Flood level tran	smiti	ter	rs.						

LICENSEE E	VENT REPORT	1 NUCLEAN ALCOLATONT CO		151 00 00 00 00 00 00 00 00 00 00 00 00 00	ATED O MATION ENTS B LATOR I APERW NAGEW	COLLE COLE CO	EXPINE EXPINE FER AN COLOR SSION DUCTION DUCTION	ME NC 115 ES 4 30 92 ESPONSE 7 DEN ESTIM T BRANCH WASHINGT DN PROJEC ET WASHI	0 00 04 500 40 11 10 1 0 00 00 1 100 0 00 00 1 100 1 100 0 00 00		
TY NAME (1)		DOCKET NUMBER (2)				NUMBER	(0)				1
CRYSTAL RIVE	R UNIT 3	0 15 10 10 10 1	31012	910	- (0 1 01	5 -	0 10	0 16	OF	016
TAG NO. SP-18-LT [JB,LT] SP-19-LT SP-21-LT SP-22-LT SP-23-LT SP-24-LT SP-25-LT SP-27-LT SP-31-LT SP-32-LT VALVES	SAFE SHUTDOW EQUIPMENT DI Emergency Fo transmitter low Steam G	N EQUIPMENT AFFE ESCRIPTION eedwater Initiat s used for initi enerator level.	CTED (ion ar ation	CONT od Co of E	.) ntro merg	ol le gency	vel Fee	edwate	r on		
CAV-1 [KN,V] CAV-3 CAV-126 "	RCS samplin	g valves and cor	tainm	ent i	sola	ation	i va'	lves.			
MUV-40 [CB,V] MUV-41 MUV-505	Letdown Coo valves.	ler isolation va	lves	and c	onta	ainme	ent	isolat	ion		