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RECIP.NAME RECIPIENT AFFILIATION

SUBJECT: LER 91-024-00:on 910909,unanalyzed condition associated w/ postulated main steam line failure outside containment discovered by GE.Caused by failure to consider iodine source term.Procedure re cold startup changed.W/911010 ltr.

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TITLE: 50.73/50.9 Licensee Event Report (LER), Incident Rpt, etc.

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WASHINGTON PUBLIC POWER SUPPLY SYSTEM

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Docket No. 50-397

October 9, 1991
G02-91-183

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

Subject: NUCLEAR PLANT NO. 2
LICENSEE EVENT REPORT NO. 91-024

Dear Sir:

Transmitted herewith is Licensee Event Report No. 91-024 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.

Very truly yours,

J. W. Baker
WNP-2 Plant Manager

Enclosure:
Licensee Event Report No. 91-024

cc: Mr. John B. Martin, NRC - Region V
Mr. C. Sorensen, NRC Resident Inspector (M/D 901A)
INPO Records Center - Atlanta, GA
Ms. Dottie Sherman, ANI
Mr. D. L. Williams, BPA (M/D 399)
NRC Resident Inspector - walk over copy

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NRC FORM 366 (6-89)

U.S. NUCLEAR REGULATORY COMMISSION

APPROVED OMB NO. 3150-0104
EXPIRES: 4/30/92

LICENSEE EVENT REPORT (LER)

ESTIMATED BURDEN PER RESPONSE TO COMPLY WTH THIS INFORMATION COLLECTION REQUEST: 50.0 HRS. FORWARD COMMENTS REGARDING BURDEN ESTIMATE TO THE RECORDS AND REPORTS MANAGEMENT BRANCH (P-530), U.S. NUCLEAR REGULATORY COMMISSION, WASHINGTON, DC 20555, AND TO THE PAPERWORK REDUCTION PROJECT (3150-0104), OFFICE OF MANAGEMENT AND BUDGET, WASHINGTON, DC 20503.

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
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TITLE (4) Unanalyzed Condition Associated With Main Steam Line Failure Outside 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 NUMBER(S)
0 9	0 9	9 1	9 1	0 2 4	0 0	1 0	0 9	9 1		0 5 0 0 0
										0 5 0 0 0

OPERATING MODE (9): 4

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)	[ ] 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 (AREA CODE)
C. L. Fies, Compliance Engineer	5 0 9 3 7 7 - 4 1 4 7

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)

[ ] YES *(If yes, complete EXPECTED SUBMISSION DATE)* [X] NO

EXPECTED SUBMISSION DATE (15): MONTH | DAY | YEAR

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

ABSTRACT

On September 9, 1991 a reportability evaluation was approved which concluded that an unanalyzed condition associated with a postulated main steam line failure outside containment had existed at WNP-2 during prior operating cycles. This condition was discovered by General Electric while performing a recalculation of the accident as a result of updated meteorological data. The unanalyzed condition was caused by the need to consider an additional iodine source term as a result of the postulated mass release through main steam line drains in the event of a high energy pipe break outside the primary containment. This unique accident scenario results in Reactor Pressure Vessel (RPV) depressurization and an increased radiological source term due to iodine spiking that is not usually associated with a main steam line break.

Since the plant was is mode four (cold shutdown) no immediate corrective action was required for the unit itself. Immediate corrective action was taken to change Plant Procedure PPM 3.1.2, Reactor Plant Cold Startup, to require the five drain valves located outside containment (MS-V-67A, B, C, D, and MS-V-19) to be closed whenever reactor power is greater than or equal to five percent.

The root cause of this event was less than adequate change management. The risks and consequences associated with the change in operating procedures was not adequately reviewed and assessed.

Further corrective actions will be taken to assure personnel are aware of the need to carefully control changes. Calculation procedures will be strengthened and a review will be performed to assure adequate controls are in place for calculations done by internal organizations.

The safety significance review showed the impact of the postulated accident, had it occurred with depressurization, would have been greatly reduced because of the very small number of failed fuel rods during past operating cycles. In addition, downstream valves, even though not safety related, would have most likely been available to stop the long term depressurization.

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

Plant Conditions
Power Level - 0 %
Plant Mode - 4

Event Description

On September 9, 1991 a reportability evaluation was approved which concluded that an unanalyzed condition associated with a postulated main steam line break (MSLB) outside containment had existed at WNP-2 during prior operating cycles. This condition was discovered by General Electric while performing a recalculation of the FSAR Chapter 15 accidents and transients as a result of updated meteorological data. The unanalyzed condition involved the amount of water and steam released in the event of a MSLB outside the primary containment. The current accident analysis, as described in Section 15.6.4 of the FSAR, assumes a break in a main steam line downstream of an outboard isolation valve. A single failure of one of the inboard Main Steam Isolation Valves (MSIV) is also assumed in the analysis. Within seconds, however, the remaining outboard MSIV closes, and the release is limited to the mass flowing through the valve while it is closing (119,000 pounds). This current FSAR accident scenario does not result in significant primary system depressurization.

NRC FORM 366A (6-89)

U.S. NUCLEAR REGULATORY COMMISSION

LICENSEE EVENT REPORT (LER) TEXT CONTINUATION

APPROVED OMB NO. 3150-0104
EXPIRES: 4/30/92

ESTIMATED BURDEN PER RESPONSE TO COMPLY WTH THIS INFORMATION COLLECTION REQUEST: 50.0 HRS. FORWARD COMMENTS REGARDING BURDEN ESTIMATE TO THE RECORDS AND REPORTS MANAGEMENT BRANCH (P-530), U.S. NUCLEAR REGULATORY COMMISSION, WASHINGTON, DC 20555, AND TO THE PAPERWORK REDUCTION PROJECT (3150-0104), OFFICE OF MANAGEMENT AND BUDGET, WASHINGTON, DC 20503.

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

The condition discovered involves additional mass release through main steam drain line piping under the above accident scenario. These drain lines are isolated outside primary containment by five main steam line drain valves (MS-V-67A, B, C, D, and MS-V-19). During heatup and cooldown conditions the drain lines that each contain a 67 valve provide a means of removing moisture from their associated main steam lines in the area between the inboard and outboard isolation MSIVs. The fifth drain line that contains MS-V-19 is designed to remove moisture from all main steam lines upstream of the inboard isolation valves. During the postulated accident conditions jet impingement from the steam line break outside containment could cause these valves to fail "as-is"at the time the accident occurred. During the past operating cycles (since early 1984) WNP-2 has been operating with these valves. open. If the main steam line break accident had occurred, as described in the FSAR, additional mass could be released through these valves. The highest mass release would occur if blowdown occurred through the line containing MS-V-19 since this is a three inch line while the lines containing the 67 valves are one and one half inch diameter lines. A single failure, plus the consequences of the jet impingement, even on all the MS-V-67 valves and the MS-V-19 valve, can only result in one unisolated blowdown path being available. The single failure is either the failure of an inboard MSIV to close (leading to blowdown through one MS-V-67 valve) or the failure of MS-V-16 (leading to blowdown through MS-V-19).

Any of these lines could result in long term depressurization of the primary system if isolation by downstream non-safety related valves could not be achieved. The depressurization could also result in iodine release from the failed fuel assumed to be present just prior to the event. This iodine release would cause "iodine spiking" usually associated with instrument line breaks in boiling water reactors and steam generator tube ruptures in pressurized water reactors. The long term depressurization aspect of the event described above makes it different from the accident described in Standard Review Plan 15.6.4, Radiological Consequences of Main Steam Line Failure Outside Containment (BWR). Consequently, this scenario would increase the radiological source term for the calculation beyond that evaluated in the FSAR resulting in an unanalyzed condition.

Immediate Corrective Action

Since the plant was in mode four (cold shutdown) no immediate corrective action was required for the unit. Immediate corrective action was taken to change Plant Procedure PPM 3.1.2, Reactor Plant Cold Startup, to require the drain valves to be closed whenever reactor power is greater than or equal to 5 percent.

Further Evaluation and Corrective Action

A. Further Evaluation

1. This event is being reported per the requirements of 10CFR50.73(a)(2)(ii)(B) as ".....a condition that was outside the design basis of the plant.....".

2. Further evaluation of this event showed a long history associated with this issue:

   a. The original design of the plant and the supporting analysis done by General Electric prior to 1983 assumed the drain valves were closed (or could be closed) under accident conditions. The Plant Operating Procedure in effect in mid 1983, PPM 3.1.2, Reactor Plant Cold Startup, Revision 4, implemented this design requirement by requiring the valves to be closed when the plant was above 5% power. Application of the rules for high energy line break analysis allow a valve to be considered "closed" if it is closed at or below 5% rated power.

   b. In December 1983 Burns and Roe, the Architect-Engineer for WNP-2, completed the stress analysis on the drains from the main steam lines and determined that they could not be certified for the life of the plant if the valves were closed at greater than 5% power and reopened at below 5% power. They recommended adding orifices to the lines and letting them blowdown continuously in order to avoid the severe thermal stress cycles. This recommendation was implemented by a change to PPM 3.1.2 (Revision 5) in April 1984. It did not appear that the personnel making the change to the procedure were aware of the requirements of the design and the accident analysis and the plant was operated with the valves open until the refueling outage in 1985.

   c. In May 1985 a nonconformance report (285-0234) was generated when it was determined that both FSAR, section 3.6, and Burns and Roe calculation 5.51.059 required the drain valves to be closed at greater than 5% power. The nonconformance report again raised a concern about the thermal stress on the valves. As a result of this concern, a evaluation was performed to evaluate the radiological

consequences of operating with the valves open. This evaluation was performed by one of the plant support organizations, Radiological Programs, at the request of Engineering. The evaluation assumed an additional 1,080,000 pounds of saturated liquid and steam was released because of the open drain valves but the radiological source term was taken from the FSAR analysis. With the assumptions used, the calculated doses were all less than one rem and, based on these results, the decision was made to operate above 5% power with the drain valves open. The plant continued to operate in this condition until the refueling outage in 1991.

d. In early 1991 General Electric was contracted to recalculate the off-site consequences of the main steam line failure outside containment accident. This was done to incorporate revised meteorology that was added to the FSAR in the 1990 amendment. When it was learned by General Electric that the reason for the high steam and liquid mass losses provided by the Supply System for this event was that WNP-2 was operating with the drain valves opened it was pointed out that the increased source term due to iodine spiking needed to be included in the evaluation. A preliminary calculation was performed by General Electric using a non-production program to estimate the dose. In this evaluation the total iodine spiking source term calculated from the instrument line break accident (WNP-2 FSAR Section 15.6) was assumed to be released to the reactor pressure vessel immediately following a main steam isolation valve closure. This is a conservative assumption since the normal spiking source term released to the RPV is proportional to the rate of vessel depressurization. Also, as the RPV depressurizes less of the mass loss will be saturated steam and, therefore, less iodine will be released. Further, since the mass of coolant lost through the drain lines is in excess of the coolant initially in the RPV it is assumed that all the spiking activity is released to the environment. With these assumptions the calculated maximum offsite thyroid dose is 239 rem which is within the 300 rem limit of 10CFR100. The evaluation by General Electric was not finalized with a production program since the decision was made to operate with the valves closed.

3. Further investigation found that the evaluation done in 1985 did not use the formal process established by Radiological Programs for performing calculations to be used for safety related applications. The memo transmitting the results of the evaluation recommended a refined analysis be performed using codes referenced in the FSAR. Radiological Programs Instruction (RPI) Manual Procedure RPI 2.1, Evaluation Log, should have been used to perform the evaluation for this application.

NRC FORM 366A (6-89)

U.S. NUCLEAR REGULATORY COMMISSION

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

4. Further evaluation did not find any record of a Generation Engineering review of the results of the evaluation. Generation Engineering had formally requested the evaluation be performed by memo and had provided the mass release value. However, since the formal calculation was not completed and transmitted to Engineering no review was performed. The only record of review was the nonconformance report which used the Radiological Programs memo as justification for closeout of the NCR. There was no Engineering signoff of the NCR as it was not required by the process at that time. Generation Engineering believed a formal calculation would be performed by Radiological Programs and delayed changing the FSAR to reflect the open valves until it was received. However, Radiological Programs was not aware of the need for a formal calculation. Tracking systems in place at that time were not sufficient to flag this item for management attention.

5. A 50.59 review of the change to Plant Procedure PPM 3.1.2, Reactor Plant Cold Startup, was performed in March 1984. The 50.59 review should have recognized the significance of the change which allowed operation with the drain valves open.

6. The root cause of this event was less than adequate change management. The risks and consequences associated with the change in operating procedures was not adequately reviewed and assessed. The change management process should have detected this deficiency in 1984 when the initial change to the procedure was made and again in 1985 when the non-conformance report was signed off with a use-as-is disposition. This event also had three contributing causes. Required procedures were not used by Radiological Programs in performing the radiological analysis for this safety related application. In addition, there was inadequate communication between organizations on the end use and status of the evaluation. Finally, personnel performance was a contributing cause. The formal calculation should have been done in a timely manner and, failing that, the issue should have been brought to management attention.

7. There were no structures, components or systems that were inoperable prior to the start of this event which contributed to the event.

B. <u>Further Corrective Action</u>

1. A lessons learned evaluation will be generated as part of the root cause analysis and this will be made a part of the training or required reading for all Plant Technical, Plant HP/Chemistry, Generation Engineering, and Radiological Programs personnel performing safety related calculations.

2. Radiological Health Instruction 2.1, Calculation Logs, will be amended to ensure the intended use of the evaluation is clear to the individual performing the evaluation and that the use is identified within the body of the evaluation.

3. An adequacy review will be performed of the present policies and processes used for calculations by internal organizations supporting WNP-2 Design Engineering.

4. The impact of thermal cycling on the drain valves and lines will be reviewed to assure design limits will not be exceeded.

5. No further corrective action is believed to be necessary for the 50.59 and change management process. The process has been strengthened significantly since this event occurred in the 1984-1985 time frame. Methods of identifing and tracking corrective actions have been improved. This includes additional training that is ongoing at this time. This should prevent reoccurrence of this problem in the future.

6. No further corrective action is needed for tracking outstanding items as this process has also been strengthened to assure periodic reviews are performed by management in a timely manner.

Safety Significance

The plant has been operating outside the bounds of the analysis as described in Chapter 15 of the FSAR. However, one factor that would mitigate the significance of this condition involves the capability for isolation using the downstream valves (see attached figure). The long term depressurization resulting from the MS-V-19 or a MS-V-67 valve failing open during a MSLB could have been prevented by closure of MS-V-21 and MS-V-156 (for MS-V-19) or MS-V-68 and MS-V-69 (for a MS-V-67 valve). Credit is not taken for these valves and associated piping in the accident analysis as they are not safety related.

An additional important factor that would decrease the significance of the MSLB, if it had occurred, would be the condition of the fuel at WNP-2 during past operating cycles. The number of fuel rod failures has been very small. The iodine spiking occurs when the gas trapped within a failed rod during power operation is released on depressurization. The accident analysis recently performed by General Electric that resulted in the calculated thyroid dose of 239 rem assumes a fuel failure fraction at the start of the accident that was typical of BWR reactor experience more than 10 years ago. At WNP-2 there have been only six rod failures since plant startup. The worst case fuel cycle involved four pin failures. Therefore, if a MSLB accident had occurred there would have been significant releases of primary coolant but the radiological consequences would have been below 10CFR100 limits because of the low amount of fuel failures.

NRC FORM 366A
(6-89)

U.S. NUCLEAR REGULATORY COMMISSION

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TEXT CONTINUATION

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Similar Events

There are no similar events.

EIIS Information

Text Reference	EIIS Reference System	Component
Main Steam System	SB	–
Main Steam Drain Valves (MS-V-67A, B, C, D, and MS-V-19)	SB	V

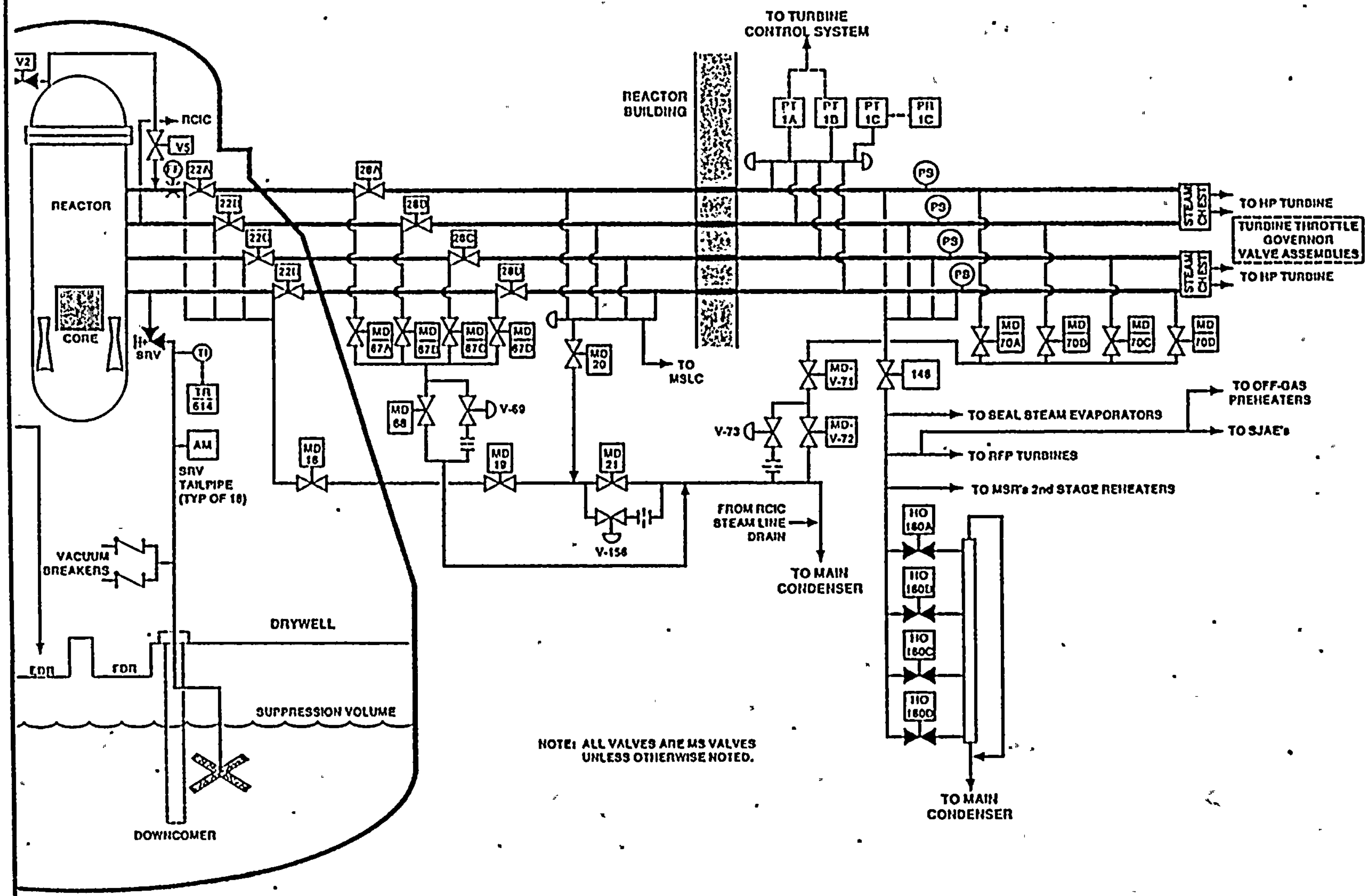

FIGURE 1. MAIN STEAM SYSTEM