WOLF CREEK
NUCLEAR OPERATING CORPORATION

Matthew W. Sunseri
Vice President Operations and Plant Manager

August 25, 2009

WO 09-0024

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

Reference: Letter WO 08-0025, dated October 3, 2008, from M. W. Sunseri, WCNOC, to the USNRC

Letter WO 09-0002, dated January 30, 2009, from M. W. Sunseri, WCNOC, to the USNRC

Subject: Docket No. 50-482: Licensee Event Report 2008-008-02, Potential for Residual Heat Removal Trains to be Inoperable during Mode Change

Gentlemen,

The references submitted Licensee Event Report (LER) 2008-008-00 and LER 2008-008-01, which described the potential for the Residual Heat Removal Trains to be inoperable during a change from Mode 4 to Mode 3. It was submitted pursuant to 10 CFR 50.73(a)(2)(i)(B) and 10 CFR 50.73(a)(2)(ii)(B).

The enclosed LER 2008-008-02 is being submitted because further evaluation was conducted which provides additional detail of the event. The event is also being submitted pursuant to 10 CFR 50.73 (a)(2)(v)(B) an event or condition that could have prevented the fulfillment of a safety function.

This letter contains no commitments. If you have any questions concerning this matter, please contact me at (620) 364-4008, or Mr. Richard D. Flannigan, Manager Regulatory Affairs at (620) 364-4117.

Sincerely,

M W Sunseri

Matthew W. Sunseri

MWS/rlt

Enclosure

cc: E. E. Collins (NRC), w/e
V. G. Gaddy (NRC), w/e
B. K. Singal (NRC), w/e
Senior Resident Inspector (NRC), w/e

IE22
NRR

NRC FORM 366 (9-2007) U.S. NUCLEAR REGULATORY COMMISSION	APPROVED BY OMB: NO. 3150-0104 EXPIRES: 08/31/2010
LICENSEE EVENT REPORT (LER) (See reverse for required number of digits/characters for each block)	Estimated burden per response to comply with this mandatory collection request: 50 hours. Reported lessons learned are incorporated into the licensing process and fed back to industry. Send comments regarding burden estimate to the Records and FOIA/Privacy Service Branch (T-5 F52), U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, or by internet e-mail to infocollects@nrc.gov, and to the Desk Officer, Office of Information and Regulatory Affairs, NEOB-10202, (3150-0104), Office of Management and Budget, Washington, DC 20503. If a means used to impose an information collection does not display a currently valid OMB control number, the NRC may not conduct or sponsor, and a person is not required to respond to, the information collection.

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4. TITLE
Potential for Residual Heat Removal Trains to be Inoperable during Mode Change

5. EVENT DATE			6. LER NUMBER			7. REPORT DATE			8. OTHER FACILITIES INVOLVED	
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REV NO.	MONTH	DAY	YEAR	FACILITY NAME	DOCKET NUMBER 05000
08	04	2008	2008	- 008 -	02	08	25	2009	FACILITY NAME	DOCKET NUMBER 05000

9. OPERATING MODE: 1

10. POWER LEVEL: 100

11. THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR§: *(Check all that apply)*

☐ 20.2201(b)	☐ 20.2203(a)(3)(i)	☐ 50.73(a)(2)(i)(C)	☐ 50.73(a)(2)(vii)
☐ 20.2201(d)	☐ 20.2203(a)(3)(ii)	☐ 50.73(a)(2)(ii)(A)	☐ 50.73(a)(2)(viii)(A)
☐ 20.2203(a)(1)	☐ 20.2203(a)(4)	☑ 50.73(a)(2)(ii)(B)	☐ 50.73(a)(2)(viii)(B)
☐ 20.2203(a)(2)(i)	☐ 50.36(c)(1)(i)(A)	☐ 50.73(a)(2)(iii)	☐ 50.73(a)(2)(ix)(A)
☐ 20.2203(a)(2)(ii)	☐ 50.36(c)(1)(ii)(A)	☐ 50.73(a)(2)(iv)(A)	☐ 50.73(a)(2)(x)
☐ 20.2203(a)(2)(iii)	☐ 50.36(c)(2)	☐ 50.73(a)(2)(v)(A)	☐ 73.71(a)(4)
☐ 20.2203(a)(2)(iv)	☐ 50.46(a)(3)(ii)	☑ 50.73(a)(2)(v)(B)	☐ 73.71(a)(5)
☐ 20.2203(a)(2)(v)	☐ 50.73(a)(2)(i)(A)	☐ 50.73(a)(2)(v)(C)	☐ OTHER Specify in Abstract below or in NRC Form 366A
☐ 20.2203(a)(2)(vi)	☑ 50.73(a)(2)(i)(B)	☐ 50.73(a)(2)(v)(D)	

12. LICENSEE CONTACT FOR THIS LER

FACILITY NAME	TELEPHONE NUMBER *(Include Area Code)*
Richard D Flannigan, Manager Regulatory Affairs	(620) 364-4117

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

CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX	CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX

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

ABSTRACT *(Limit to 1400 spaces, i.e., approximately 15 single-spaced typewritten lines)*

On August 4, 2008, during a review for potential void formation to address Generic Letter 2008-01, a concern with the Residual Heat Removal (RHR) system during Mode 4 and Mode 3 was identified.

During shutdown conditions, Wolf Creek Generating Station typically lines up one or both of the RHR trains to the Reactor Coolant System (RCS) for shutdown cooling and switches them to RHR Injection mode of operation prior to entering Mode 3 from Mode 4. During the switchover, system procedures require the RHR system to be cooled down, using the mini-flow line through the RHR heat exchanger. The physical location where the mini-flow piping returns to the suction of the RHR system prevents cooling approximately 140 feet of the RHR suction line. The stagnant water in the RHR suction line cools by ambient losses only and can remain high (maximum 350 degrees F) for several hours. The saturation pressure for this hot water can prevent the check valve from the Refueling Water Storage Tank from opening, preventing flow to the suction of the RHR pumps during a Mode 3 Loss of Coolant Accident.

The condition with hot water in the RHR suction piping can also occur shortly after initiating RHR Shutdown cooling mode of operation during the cooldown and would exist until RCS temperatures are reduced sufficiently to preclude a concern. Further review of the plant conditions during Refueling Outage 16 showed that not only was the RHR system inoperable during the heat up at the end of the outage, but it also occurred during cool down at the beginning of the outage.

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PLANT CONDITIONS PRIOR TO EVENT:

MODE – 1
Power – 100

EVENT DESCRIPTION:

On August 4, 2008, during a review for potential void formation to address Generic Letter 2008-01, a concern with the Residual Heat Removal (RHR) system [EIIS Code: BP] during Mode 4 and Mode 3 was identified.

During shutdown conditions, Wolf Creek Generating Station (WCGS) typically lines up one or both of the RHR trains to the Reactor Coolant System (RCS) [EIIS Code: AB] for shutdown cooling and switches them to RHR Injection mode of operation prior to entering Mode 3 from Mode 4. System procedures require the RHR system to be cooled down using the mini-flow recirculation line, following alignment for Emergency Core Cooling System (ECCS) injection mode. The location where the mini-flow piping returns to the suction of the RHR system prevents approximately 140 feet of RHR suction line between the RCS hot leg isolation valve and the mini-flow line location from being cooled.

If the RHR system is aligned to ECCS injection Mode with water temperature near 350 degrees F, the water in the RHR suction piping will remain hot for a considerable duration. If a Mode 3 Loss of Coolant Accident (LOCA) were to occur and Safety Injection System (SIS) initiated [EIIS Code: JE], the RHR pump would start, resulting in lowering the pressure in the suction piping. This lowering of pressure will result in flashing the hot water into steam, thereby voiding the RHR suction piping. As long as the saturation pressure in the RHR suction leg is higher than the static pressure from the Refueling Water Storage Tank (RWST), the check valve located in the supply line from the RWST, will not open and no injection from the RWST will occur. The steam void could extend to the pump suction and steam bind the pump.

A review of plant conditions during the startup of WCGS from Refuel Outage 16 showed that the condition existed when changing from Mode 4 to Mode 3 on May 10, 2008. As a result, RHR was not operable as required per Technical Specification 3.5.2 and 3.5.3.

During the investigation, a related potential for water hammer was also identified whereby the steam voided RHR piping would subsequently refill from either the RWST during the ECCS injection mode of operation, or the containment recirculation sump during the ECCS recirculation mode of operation. This water hammer has the potential to damage RHR system piping, components, and supports.

The condition with hot water in the RHR suction piping can also occur shortly after initiating RHR Shutdown cooling mode of operation during the cooldown and would exist until RCS temperatures are reduced sufficiently to preclude a concern. Further evaluation revealed that an RHR system was also inoperable during the cool down at the beginning of Refueling Outage 16. In Mode 3, both RHR trains are maintained in the injection mode with the suction lined up to the RWST. During a cool down, neither train is limited by suction temperature in Mode 3, as they are both near ambient conditions. However, entry into mode 4 involves aligning one or both RHR subsystems for Shutdown Cooling (SDC) mode of operation with the suction lined up to the RCS to complete the cool down of the RCS.

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Cooled RCS water is returned to the RCS through the RHR Heat Exchanger. Since Mode 4 begins at an RCS temperature of 350 degrees F, the RHR suction may be at that temperature. The time of concern is during the upper temperatures of Mode 4 during the cool down. Of significance, an RHR subsystem lined up for SDC with the RCS at high temperature may not be available for injection mode of operation should it be required. This is contrary to Technical Specification (TS) 3.5.3, which requires the RHR system to be capable of being manually realigned to the ECCS mode of operation. During the evaluation, it was found that these conditions existed in previous Refueling Outages.

In addition, had this condition been recognized both during the cooldown and the heatup, TS 3.0.3 and 3.0.4 should have been entered.

BASIS FOR REPORTABILITY:

Wolf Creek changed from Mode 4 to Mode 3 at the end of RF16 without ensuring the RHR system was operable. In addition, Wolf Creek changed from Mode 3 to Mode 4 at the beginning of RF16 without ensuring that the RHR system was operable. These events are reportable under 10 CFR 50.73(a)(2)(i)(B) as operation or condition prohibited by Technical Specifications, under 10 CFR 50.73(a)(2)(ii)(B) for being in an unanalyzed condition, and under 10 CFR 50.73 (a)(2)(v)(B) as a condition that could have prevented the fulfillment of a safety function.

ROOT CAUSE:

The root cause of the failure to ensure RHR operability is the RHR system design was not adequate to support all modes of RHR operation without impacting each other. Contributing cause includes failure to adequately evaluate NSAL 93-004, RHR System Operation as Part of the ECCS During Plant Startup.

CORRECTIVE ACTIONS:

Operations procedures were revised to ensure, when RHR is aligned to ECCS injection mode, that the temperature of the RHR suction lines from the RCS isolation valve are within prescribed limits.

SAFETY SIGNIFICANCE:

Short Term Core Cooling - Injection Phase Small Break LOCA

A conservative thermal-hydraulic analysis of a 6-inch equivalent diameter small break LOCA has been performed to demonstrate the relatively benign characteristics of shut-down LOCAs under these circumstances. This analysis used the NRC approved small break LOCA Evaluation Model with the NOTRUMP computer code.

The results of the analysis provides reasonable assurance that the limits of 10CFR50.46 will not be exceeded if a small break LOCA occurs in Mode 3 or 4, provided the operator establishes flow from a high head centrifugal charging pump (CCP) within 10 minutes after event initiation. Break sizes smaller than 6 inches may require additional operator actions to depressurize the RCS or start additional safety injection (SI) pumps. However, more time is available for the operator to respond to accomplish the required actions and current procedures direct such

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activities. Therefore, it is concluded that a small break LOCA occurring in Mode 3 or Mode 4 can be successfully mitigated, if flow is available from one high head charging/SI pump in a timely manner, and that the RHR pumps need not be relied on for short-term LOCA consequences mitigation. As a result, a loss of the RHR injection capability during shutdown conditions would pose insignificant consequences.

A cladding heat-up calculation was performed based on bounding RCS transient response. As the RCS response is considered to be overly conservative, a more realistic shut-down time (8 hours) and decay heat model, per ANS-5.1-1979, were assumed as the basis for the specific fuel rod heat flux. The 8 hour shut-down time is reflective of the forced outage that occurred shortly after the Refuel Outage 12 outage. The results are shown below:

Peak Cladding Temperature (PCT), 1,234.8 degrees F
PCT Time, 1,810.4 sec
PCT Elevation, 11.75 ft
Maximum Hot Rod ZrO2, 0.16 percent Equivalent Cladding Reacted (ECR)
Maximum Hot Rod ZrO2 Elev., 11.75 ft
Hot Rod Axial Average ZrO2, 0.02 percent ECR

In addition, results from the 6" Normal Operating Pressure/Normal Operating Temperature (NOP/NOT) case demonstrate that RHR is not required for short-term core cooling assuming both RHR pumps fail to start. Core uncovery is brief and core exit vapor temperature never exceeds initial, sub-cooled coolant starting temperatures.

Although smaller break sizes were not analyzed, the results from these break sizes are considered to be less limiting since the amount of mass loss with respect to time would be reduced, while make-up capability would remain similar to that of the 6 inch case. This is due to the performance characteristics of the CCP at lower pressures (relatively constant), which is where these transients trend due to their low power aspects.

Based on this heat-up calculation, it is concluded that for breaks up to and including a 6" double-ended severance of a reactor coolant pipe during shutdown conditions, the remaining ECCS subsystem limits the peak cladding temperature well below the10 CFR 50.46 limit of 2200 degrees F and ensures that the core remains in place and substantially intact with its essential heat transfer geometry preserved.

Short Term Core Cooling - Injection Phase Large Break LOCA Evaluation

It should be noted that a large break LOCA was not explicitly considered in Mode 3 or 4. The justification is based upon the probabilistic risk assessment (PRA) study documented in WCAP-12476 and is summarized as follows:

As documented in WCAP-12476, a PRA approach using the Structural Reliability and Risk Assessment (SRRA) methodology has been performed to assess the risk of core damage in the shutdown modes for a large break LOCA. A large break LOCA is defined here to be breaks in piping larger than 6 inches in diameter. The PRA has shown that, even with the increased reliance on operator actions and the reduction of available safety systems, the risk of core damage resulting from a large break LOCA is less in Mode 3 and Mode 4 than for Mode 1.

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The SRRA analysis is a proven and accepted analytical tool that utilizes conventional structural analysis techniques in combination with probabilistic methods to evaluate the potential for component structural failure. The SRRA analysis indicates that the probability of a large pipe break occurring at Mode 3 (hot standby when the accumulators are blocked) or Mode 4 (hot shutdown) conditions is significantly lower (by more than an order of magnitude) than at Mode 1 conditions. Given that the probability of a large break occurring at any conditions is quite small, and that the piping has had the proof test of full pressure (and temperature) integrity for the limiting scenarios, the possibility of a large break occurring during shutdown is extremely remote.

To compare the risk of core damage in Mode 1 to the risk of core damage in the shutdown modes for a large break LOCA, a probabilistic risk assessment, which incorporated the SRRA results, was then performed. The frequency of core damage in each mode was calculated as the product of the probability of a pipe break, the fraction of time in the mode of operation, and the conditional probability that a large break LOCA would not be mitigated. The relative risk was calculated as the ratio of the frequency of core damage in either Mode 3 or 4 to Mode 1. This resulted in mean relative risk probabilities of 1/34 for Mode 3 (with the accumulators isolated) and 1/6.9 for Mode 4, when compared to the risk for large break LOCAs during Mode 1. The lower risk posed by a large break LOCA in Modes 3 and 4 provides justification that a thermal-hydraulic analysis of a large break LOCA is not warranted.

Short Term Core Cooling - Containment Response

A previous containment pressure response analysis, based upon the mass and energy releases for the bounding 6" break small break LOCA scenario mentioned above, predicted that the containment pressure was expected to reach the Hi-3 containment pressure setpoint of 27 psig, for automatic containment spray actuation, at approximately 3.5 hours after event initiation, with only one train of containment fan coolers assumed to be operating. As a result, the RWST water inventory may be depleted soon after the containment spray pumps start to draw water from the tank.

However, it should be noted that the bounding small break LOCA mentioned above, assumed the event being initiated 2-hours after plant shutdown and used Appendix K assumptions such as decay heat, break flow, etc. If the analysis is performed with actual Mode 3 or 4 conditions, with realistic yet conservative models for decay heat (e.g., 1979 ANS standard) and break flow, the mass and energy releases are anticipated to be relatively mild, due to a lower decay heat level along with lower RCS pressures and temperatures, and consequently, the containment pressure excursion would not be dramatic.

Based on the mass and energy releases generated with realistic yet conservative models for decay heat and break flow, along with containment fan coolers at full capacity, a containment pressure response analysis has confirmed that the containment response to those credible shut-down LOCA scenarios established in this investigation is relatively quiescent. That is, the resulting containment pressure due to a credible shutdown small break LOCA may not reach the Hi-3 containment pressure (i.e., 27 psig), for automatic actuation of the containment spray. As a result, the use of the RWST water inventory for continuous safety injection could be prolonged. As such, containment integrity and the effect of the containment spray system on RWST inventory is not a factor in the over-all safety significance of this issue.

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Long Term Core Cooling - Recirculation Phase LOCA Evaluation

Plant specific evaluations conclude that RHR recirculation capability is not lost due to the effects of a potential condensate induced water hammer. The scenarios involve conditions where elevated temperature water remains in the RHR suction line at the time of realignment to containment sumps because the pumps do not inject into the RCS or they remain in stand-by. For an small break LOCA of 1.687 inch equivalent diameter or less, RCS pressure will remain above the RHR shut-off pressure and preclude injection.

With respect to the potential loss of RHR capability during recirculation after successfully mitigating a small break LOCA greater than 1.687 inch, a qualitative assessment is provided as follows:

Should the RHR pump and/or system be damaged beyond its intended design basis function as a result of steam binding during a postulated LOCA, the operating CCP and/or SI pump would have no suction source from the RHR pump discharge for ECCS recirculation. However, the minimum ECCS flow requirements for removing the decay heat in the core can still be met. This is accomplished through the use of guidance provided in Emergency Procedure EMG C-11, Loss of Emergency Coolant Recirculation. This procedure has been in place at Wolf Creek during the time frames in which the RHR system was vulnerable to suction line elevated temperature scenarios. It is intended to mitigate situations where sump recirculation is not available. In the vulnerability time frames of interest, this procedure would be used in any of a number of instances including failure of an RHR pump, complete sump blockage, suction valve failure, etc. As such, the operations staff is periodically trained to use this procedure and is familiar with the strategies it employs and the rationale behind those strategies. The key strategies in this procedure, among others, include: RWST water conservation, reducing RCS pressure to reduce system subcooling and mass loss, use of supplemental make-up sources and reflux cooling which condenses vapor within the RCS. The following discusses each of these in more detail:

RWST water conservation: When the operators diagnose that sump recirculation is not available and EMG C-11 is entered, one of the first objectives is to conserve RWST water. This is accomplished in several ways; 1) cooldown and depressurize the RCS to reduce break flow, 2) reduce or secure containment spray if conditions allow, and 3) reduce ECCS flow to a single train.

In the next sequence of steps the operators will determine the minimum amount of ECCS flow required for core cooling in order to conserve what inventory remains in the RWST. In the case of the shut-down LOCAs considered here, one train of ECCS would allow operation of the RWST down to 36% in approximately 3.3 hours. When the RWST low-low-1 water level of 36% is reached, the minimum ECCS flow required would be employed through the use of; flow estimates considering shut-down time specified in the procedure the reactor vessel level indicating system (RVLIS) and trends on core exit thermo-couple readings. This would extend the operation of the ECCS on the RWST for several more hours until RWST water level reaches 6%. At this point, the operators are instructed to align one CCP to the volume control tank (VCT), establish normal charging flow and check chemical control volume system (CVCS) make-up capability. Adequate RCS makeup is determined by maintaining RVLIS level above the top of the core and monitoring core exit thermocouples.

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Alternate make-up sources: When the RWST level reaches 6% or less, alternate makeup sources to the ECCS are utilized in EMG C-11. This includes the normal make-up system and the fire suppression system. When RWST level drops below 6 percent, ECCS flow is suspended. At this point, one charging pump is aligned to the volume control tank (VCT), with flow delivered to the RCS through this path. During this time, checks are made for adequate ECCS flow using the methods discussed above. If supplemental flow is required during this time, RWST inventory is replenished with the fire suppression system. Note: The make-up capability to the volume control tank (VCT) is approximately 120 gpm. However, if the time after shut-down is less than approximately 60 hours, 120 gpm of make-up could be insufficient to offset the core boil-off steaming rate. A strong mechanism exists to compensate for periods in which makeup alone is insufficient by dumping steam as necessary to maintain a stable or decreasing RCS temperature. This mechanism employs using the steam generators to promote reflux condensation. In addition, another condensation mechanism is present which helps with this deficit. This process is the condensation of vapor that interacts with incoming ECCS water.

Reflux Cooling: Reflux condensation is a mechanism for core cooling that can be utilized for loss of normal shut-down cooling events. However, it can also be a significant mitigating factor for small break LOCA events. In fact, it is an integral mitigating strategy credited throughout the Emergency Operating Procedure (EOP) network whenever RCS cool down is required and forced or natural circulation cannot be not established. In the reflux condensation cooling mode, core decay heat is removed by boiling; the steam flows to the steam generators where it is condensed on the inner surfaces of the steam generator (SG) tubes. The condensate from the upflow side of the U-tubes flow downward, against the upward flowing steam, into the SG inlet plenum, hot leg, and reactor vessel upper plenum, before returning to the core. The condensate from the downflow side of the U-tubes is returned to the cold leg. To be viable, the reflux cooling mode requires that one or more steam generators be operational. In these situations, the secondary-side heat sink is maintained by dumping steam to the main condenser or atmosphere with secondary side inventory provided by the auxiliary feedwater system. This mechanism has been proven to exist and can be used to compensate for the make-up short-fall that is of interest here.

Latter steps in EMG C-11 instruct the operators to depressurize all intact steam generators to atmospheric pressure. This reduces the temperature of the secondary side to a relatively low value. At this point, the steam is drawn into the SGs, since the condensation process creates a low pressure zone in these areas. As long as the temperature of secondary remains below that of the primary, this process will continue. If off-site power is available, the SG secondary side temperature can be dropped to sub-atmospheric conditions since the main condenser would be in service. This further reduces the secondary side temperature, extending the process. Considering the plausible small break sizes of interest here, a temperature difference of least 50 – 80 degrees F can be expected between the primary and secondary side. When coupled with almost 220,000 square feet of heat transfer area available in the steam generators, the condensation potential is extremely large. Thus, even with non-condensable gases present, this condensation process has the ability to more than off-set any deficit between core boil-off and the normal make-up system.

ECCS condensation also has a role in reducing make-up requirements. In simple terms this process is based on interfacial heat transfer between the injected ECCS flow and vapor present in the cold legs. Since the ECCS flow is sub-cooled, water droplets in the injection flow can and will remove the latent heat present in the vapor. Some of this vapor will then condense and remain within the RCS, that may otherwise be removed through the break.

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Note that potential reliance on reflux and ECCS condensation would only occur for LOCA's after the plant has been tripped, and is in Mode 3/4 cooling down towards a cold shut-down status. It is not needed for mitigation success in start-up LOCA scenarios. This is because during Mode 3/4 start-up, the decay heat from the core is very low. As such, the normal make-up system has more than sufficient capacity for these hypothesized scenarios under start-up conditions.

Containment Flooding: Employing the EMG C-11 strategy may ultimately involve transferring more than one RWST deliverable volume into containment. The actual weight of the additional water is not of issue since the head it creates on the building sides and floor is much less than the design pressure of the structure. However, this could create a situation, which could compromise instrumentation located in the lower compartments of the structure. This instrumentation could include vibration monitoring, reactor coolant pump (RCP) shaft speed sensors, hot and cold leg temperature indications, etc. With regard to critical instrumentation, the only mandatory information required throughout these hypothesized events is the core exit temperature monitoring system. This system is designed with appropriate environmental considerations and is located at an elevation above a potential flood level in containment that would result from these scenarios. Therefore, the system would still perform its function under these circumstances. In addition, guidelines exist that list the affected components by elevation, to allow the plant staff to determine the effect(s) due to the absence of these components, on the overall mitigation of the postulated event.

Long Term Core Cooling - Additional Recirculation Measures

RHR Pump Repair/Replacement: Should the RHR pumps become damaged and not able to be used immediately for ECCS sump recirculation, measures would be taken to repair a pump. Since the RHR system is not needed for short-term core cooling in credible shut-down LOCA scenarios, and EMG C-11 provides sufficient long term cooling strategies, ample time exists to repair a damaged RHR pump. Because of this time allowance, significant resources would be become available to accomplish this. The necessary pump spare parts exist on site or at the sister plant, such as: motor, impeller, seal package, etc. Only one pump would be needed if a repair effort is initiated.

Should repair not be an option, an entire pump could be brought to the site and installed.

Long Term Core Cooling - Alternative Recirculation Strategies

Containment Spray Pump Test Line: The containment spray pumps at Wolf Creek draw water directly from the sump during recirculation. Since the spray function is not required for the shut-down LOCAs of interest here, the spray pump(s) could be used as means for establishing a limited recirculation capability until RHR pump repairs are complete. The containment spray pump test line allows for a flow path to refill the RWST. The flow through this line is approximately 300 gpm. Because no fuel failure is expected for LOCAs of this benign nature, this method could be employed to draft water from the sump and supply it back to the top of the RWST. Latent heat from core boil-off would be removed by the containment fan coolers. As an alternative, provisions could be made to divert containment spray pump flow from the sump into the RHR system upstream of the heat exchanger. These alternate flow paths could be evaluated and implemented in a timely manner.

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Safety Significance Conclusion

If a shut-down LOCA, with a break size of 1.687 inch equivalent diameter or less, occurs at the WCGS while elevated temperature water is present in the RHR suction lines, the ECCS will be able to perform all of it's intended functions. For credible break sizes between 1.687 inch and 6 inch, RHR system functions may be compromised. However, it has been demonstrated that core damage will not occur and several means exist for long term core cooling.

Best Estimate Studies

As part of the investigation, Best-Estimate studies and analyses were performed in an effort to further understand the potential risks involved. These included determination of the RHR suction temperatures of concern, system response (pressure and stress) to potential water hammer, both from refill from RWST and refill from the containment sump, and beneficial effects of RHR mini-flow recirculation during the pump start transient. A brief summary of the results of these studies and analyses are provided below:

The temperatures of concern were shown to be above 259 degrees F for RHR pump voiding concerns and above 233 degrees F for water hammer concerns. Starting with a temperature above 340 degrees F, it would take over a day to cool the RHR piping below these temperatures with losses to ambient.

For those intervals where RHR suction piping temperature is above 259 degrees F (for RHR pump voiding concerns) or 233 degrees F (for water hammer concerns) in Modes 3 or 4:

For the Small Break LOCA (less than 2 inch schedule 160 pipe break), SI and CCP injects and there are two possibilities:

1) Starting at normal RCS operating temperature and pressure, recirculation phase is reached prior to RCS pressure falling below the RHR shutoff head, and RHR injection is precluded. Once the Recirculation phase is initiated and the RHR suction piping depressurizes to the containment sump it then refills with water from the containment sump. For those events where RHR injection flow is not required, adequate Net Positive Suction Head (NPSH) would be continuously supplied to the RHR pump suction and the RHR pump would have adequately performed its safety function in the subsequent Recirculation Phase of operation.

2) Starting at lower RCS temperatures and pressures, RHR might inject to the RCS before the recirculation phase is reached, but injection flowrate increases slowly. The cooler RHR mini-flow recirculation flow provides sufficient cooling by mixing before the RHR pump suction to assure adequate NPSH would be continuously supplied during operation, and the RHR pump would have performed its safety function in both the Injection Phase and Recirculation Phase of operation.

For Intermediate Break LOCA there is a possibility that adequate NPSH would be continuously supplied and the RHR pump will not be subjected to a voided condition provided the RHR injection flow characteristics are such that:

1) The hot water in the RHR suctioning piping is evacuated prior to closure of the min-flow recirculation valve, and

LICENSEE EVENT REPORT (LER)

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2) The suction piping water has undergone some period of cooldown before the event such that the mini-flow recirculation cooling is effective in cooling all hot water as it is being evacuated.

If these conditions are met, the RHR pump may continue functioning in the Injection Phase and the Recirculation Phase of operation. Otherwise, continued function in the Injection Phase and Recirculation Phase of operation cannot be assured for the intermediate break LOCA.

For Large Break LOCA, saturated water would have reached the RHR pump after the mini-flow recirculation valve closes, resulting in loss of NPSH. Continued function of the RHR subsystems in the Injection Phase and Recirculation Phase of operation could not be assured.

Assuming the pump remained functional throughout the injection phase, regardless of whether it injected or not, the RHR pump would not be adversely affected by subsequent switchover and operation in the Recirculation Phase.

If RHR suction piping water temperature is above 233 degrees F a water hammer may occur under certain conditions as voids are refilled in the suction piping in either the Injection Phase or in the Recirculation Phase of ECCS operation. Should a water hammer result, three piping supports may fail completely, and two may be permanently deformed. The piping may undergo limited plastic deformation. However, the piping, pump, and valves are shown to maintain their integrity regardless of the support damage and will perform their function as needed.

Risk Significance:

The risk significance herein is based on initiating events that did not occur. The basis for the risk assessment is the conditional argument that given a LOCA, equipment function is in question. The initiators are Large, Medium, and Small LOCAs, and Loss of Inventory in Mode 4. No actual equipment failures occurred. Large early release frequency (LERF) was not evaluated because the plant conditions are not conducive for generating an offsite impact.

The final risk metric of this evaluation is Incremental Conditional Core Damage Probability (ICCDP) encompassing a period of one year. A refueling outage and a forced outage occurred in 2002, which yielded multiple opportunities to expose the RHR system to the conditions being evaluated.

The total Incremental Core Conditional Damage Probability (ICCDP) is reported to be less than 1.0E-06. Changes in various parameters can move the sum total from 3.3E-07 to 4.6E-07 ICCDP. ICCDPs less than 1E-06 are considered low risk significant.

OPERATING EXPERIENCE/PREVIOUS EVENTS

None