Omaha Public Power District
1623 Harney Omaha, Nebraska 68102
402/536-4000

September 21, 1987
LIC-87-593

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

References:
1. Docket No. 50-285
2. Generic Letter 87-12, "Loss of Residual Heat Removal (RHR) While the Reactor Coolant System (RCS) is Partially Filled", dated July 9, 1987

Gentlemen:

SUBJECT: Response to NRC Generic Letter 87-12

Omaha Public Power District (OPPD) received Reference 2 on July 21, 1987. The subject Generic Letter requested a written reply, submitted under oath or affirmation pursuant to the provisions of 10 CFR 50.54(f), within 60 days of the date of receipt.

As requested and discussed with the NRC Project Manager, Mr. Anthony Bournia, OPPD is submitting the enclosed response to all items except 5 and 9. The response to item 5 is being prepared in conjunction with information recently received from the Combustion Engineering Owners Group (CEOG). As requested by Item 9, any proposed changes to current procedures will be evaluated and submitted with the response to Item 5. OPPD will provide a schedule for submittal of the response to items 5 and 9 by November 16, 1987.

OPPD plans to have any needed procedure changes in effect prior to the scheduled 1988 outage. The 1988 outage will be the next expected time when the reactor coolant system will be partially filled and reliance on the residual heat removal (RHR) system is required. If it is necessary to reduce the RCS level and use RHR prior to the 1988 outage, consideration will be given to the concerns expressed in Generic Letter 87-12. If you have any questions, please contact us.

Sincerely,

R. L. Jaworski for

R. L. Andrews
Division Manager
Nuclear Production

RLA/me
cc: LeBoeuf, Lamb, Leiby & MacRae
R. D. Martin, Regional Administrator
A. Bournia, NRC Project Manager

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UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

In the Matter of	)	
	)	
Omaha Public Power District	)	Docket No. 50-285
(Fort Calhoun Station	)	
Unit No. 1)	)	

AFFIDAVIT

R. L. Jaworski, being duly sworn, hereby deposes and says that he is the Section Manager - Technical Services, Nuclear Production Division of the Omaha Public Power District; that as such he is duly authorized to sign and file with the Nuclear Regulatory Commission the attached information relative to NRC Generic Letter 87-12, concerning the loss of residual heat removal while the reactor coolant system is partially filled; that he is familiar with the content thereof; and that the matters set forth therein are true and correct to the best of his knowledge, information and belief.

R. L. Jaworski

R. L. Jaworski
Section Manager
Technical Services

STATE OF NEBRASKA)
) ss
COUNTY OF DOUGLAS)

Subscribed and sworn to before me, a Notary Public in and for the State of Nebraska on this 21 day of September, 1987.

Melva L. Evans
Notary Public

GENERAL NOTARY - State of Nebraska
MELVA L. EVANS
My Comm. Exp. 2-11-88

Enclosure

OPPD Responses to Generic Letter 87-12

Reference 2 requested that OPPD provide responses to the following items:

Item 1

A detailed description of the circumstances and conditions under which your plant would be entered into and brought through a draindown process and operated with the RCS partially filled, including any interlocks that could cause a disturbance to the system. Examples of the type of information required are the time between full-power operation and reaching a partially filled conditions (used to determine decay heat loads); requirements for minimum steam generator (SG) levels; changes in the status of equipment for maintenance and testing and coordination of such operations while the RCS is partially filled; restrictions regarding testing, operations, and maintenance that could perturb the nuclear steam supply system (NSSS); ability of the RCS to withstand pressurization if the reactor vessel head and steam generator manway are in place; requirements pertaining to isolation of containment; the time required to replace the equipment hatch should replacement be necessary; and requirements pertinent to reestablishing the integrity of the RCS pressure boundary.

Response

The Fort Calhoun Station would enter into a Reactor Coolant System (RCS) draindown in preparation to open the RCS for maintenance and/or refueling activities. The plant is first brought to the cold shutdown condition under the procedure OI-RC-4. The Shutdown Cooling System (SDC) is placed in service per procedure OI-SC-1 once the RCS temperature and pressure are below 300°F and 250 psia respectively. With RCS pressure being maintained between 250 psia and 215 psia, the secondary side of the steam generators are drained, one at a time, and refilled with colder water to aid in cooling of the steam generators and for secondary side chemistry control. There is no Technical Specification requirement to maintain a minimum steam generator level when the RCS temperature is below 300°F, although typically the steam generator is filled to an indicated level of 100% plus 10,000 gallons. The pressurizer steam bubble is then collapsed and the RCS pressure is dropped to atmospheric pressure.

The controlling procedure for accomplishing the RCS draindown is OI-RC-5. The plant conditions prior to the initial draindown are as follows. The RCS temperature is between 130°F and 82°F. The Shutdown Cooling System is in operation in accordance with procedure OI-SC-1. The refueling level indicator, LI-197, is operable and ready to be valved in. All control element assemblies (CEAs) are fully inserted. The following pumps are verified disabled by ensuring their control switches are in the pull-stop position: the Reactor Coolant Pumps, the High Pressure Safety Injection (HPSI) Pumps, and the Charging Pumps. The pressurizer heaters are verified off.

The initial RCS draindown process begins by dropping the RCS pressure to atmospheric pressure and venting the RCS to the containment. The venting is accomplished by the removal of a blank flange located downstream of the 3/4 inch pressurizer vent valves, RC-102 and RC-167, and then opening these valves. A valve

line up is then performed and draining commences. Prior to losing level indication on the pressurizer wide range level indicator, LI-106, the refueling level indication which consists of LI-197 for remote indication in the control room and a cygon tube for local level indication in the containment is made operational. The refueling level instrumentation is described in detail under the response to Item 2. The draindown would continue until level is below the steam generator tube sheets to allow for draining of the tubes. The RCS level would be maintained at or just above the centerline elevation of the hot legs until the RCS level has stabilized, indicating the steam generator tubes have been drained.

The time between the commencement of power reduction and reaching a partially filled condition in the RCS was determined by reviewing the control room logs for each outage in which a draindown occurred in the last ten years. The smallest time interval occurred during a mid-cycle shutdown to replace two reactor coolant pump seals in December 1979. During this outage, the recorded elapsed time between the commencement of power reduction to the initiation of RCS draining was 44 hours. On this occasion, the draining process was interrupted for other concerns, but it is reasonable to conclude that the level could have reached mid-loop within 48 hours after full power operation.

Subsequent RCS draindowns within a refueling outage are usually done under different conditions from those of the initial draining for an outage. Examples of other conditions are: 1) the RCS may be vented by removal of the pressurizer safety valves and/or the removal of the Control Element Assembly uncoupling tool access flanges; 2) the reactor vessel head may be removed; 3) the steam generator nozzle dams may be in place. The RCS level instrumentation is available under these other conditions.

The RCS is designed to be operated at mid-loop for reactor coolant pump maintenance and steam generator primary side maintenance and testing. The procedures utilized are OI-RC-5 for draining and OI-RC-2A for filling of the RCS. Level is maintained during partially filled conditions by utilizing both of the above procedures. Procedure OI-RC-5 contains the precaution to maintain the RCS level above the middle of the hot leg to insure suction is available to the low pressure safety injection (LPSI) pumps that are utilized for shutdown cooling recirculation. Procedure OI-SC-1 describes the process for placing the shutdown cooling (SDC) system in service and its continued operation. This procedure specifies the SDC flow rate be set at 1500 gpm, but there is no Technical Specification requirement for a minimum flow rate.

The shutdown cooling suction piping at Fort Calhoun takes suction from the bottom of a hot leg that has a centerline elevation at 1006'-4 1/2". The suction line's containment penetration is at an elevation of 1001'. The LPSI pumps centerline is at an elevation of 973'-3". There are no loop seals in this suction line from the hot leg to the "B" LPSI pump and one loop seal, with a minimum elevation of 972'-11", to the "A" LPSI pump. Gravity feed from the refueling water storage tank to the RCS is impossible at Fort Calhoun because the maximum level, with a full storage tank, is at an elevation of approximately 1004' to 1005'. However, the LPSI pumps can take suction from this tank by gravity feed, with no loop seals in the suction lines. Should the LPSI pumps become air bound, the only venting necessary would be in the vicinity of these pumps,

and a pump could then be aligned to take suction from the refueling water storage tank and provide makeup to the RCS. Any one of the three containment spray pumps can be aligned for use as a shutdown cooling pump via procedure OI-SC-3. Fort Calhoun also has the capability of providing hot leg injection through the auxiliary pressurizer spray line by utilizing either one to three charging pumps or one HPSI pump.

The Fort Calhoun Technical Specifications require two decay heat removal loops to be operable and one decay heat removal loop to be in operation when there is less than 15 feet of water over the core. Restrictions regarding testing, operating, and maintenance are identified in each of the individual procedures that govern each respective activity. Coordination of these activities is provided through scheduling, but the ultimate responsibility of ensuring compliance with the Technical Specifications lies with the Shift Supervisor and his licensed operating staff.

The RCS can withstand pressurization up to a maximum defined by Figures 2-1A and 2-1B in the Technical Specifications without detrimental effects. This maximum allowed RCS pressure varies with temperature and reactor operating age (i.e., reactor vessel fluence). The current RCS pressure limit for temperatures between 82°F and 182°F is 545 psia. The Low Temperature Overpressure Protection (LTOP) actuation setpoint is more conservative than the maximum permissible RCS pressure and is expressed in Figure III-25 in the Fort Calhoun Operating Manual Technical Data Book (TDB). The current LTOP actuation setpoint, taken from Figure III-25, is approximately 240 psia for RCS temperatures below 130°F. The Power Operated Relief Valves open to relieve RCS pressure upon receiving an LTOP actuation signal. The containment isolation valves (HCV-347 and HCV-348) on the SDC suction line have an interlock to close when the RCS pressure is above 250 psia.

Containment closure requirements are discussed in detail in our response to Item (4). There are no Technical Specification requirements for containment closure that are dependent upon RCS level. The only time limit specified for containment closure requires all containment penetrations providing direct access from containment atmosphere to the outside atmosphere (equipment hatch included) be closed within four hours should all decay heat removal loops become inoperable when in the Refueling Shutdown mode. If necessary, the equipment hatch could be closed and secured with four bolts in approximately one hour after the work crew arrives at the equipment hatch location.

Since the condition of the RCS pressure boundary cannot be anticipated, there are no specific procedure requirements for reestablishing the RCS pressure boundary. However, the RCS pressure boundary could be reestablished provided only the closure of valves is involved.

Item 2

A detailed description of the instrumentation and alarms provided to the operators for controlling thermal and hydraulic aspects of the NSSS during operation with the RCS partially filled. You should described temporary connections, piping, and instrumentation used for this RCS condition and the quality control process to ensure proper functioning of such connections, piping and instrumentation, including assurance that they do not contribute to loss of RCS inventory or otherwise lead to perturbation of the NSSS while the RCS is partially filled. You should also provide a description of your ability to monitor RCS pressure, temperature, and level after the RHR function may be lost.

Response

Instrumentation available to the operators for maintaining residual heat removal capabilities with the RCS partially filled consists of the following. LI-197, the Refueling Level Indicator, provides level indication in the control room for RCS levels that are between the bottom of the pressurizer to the bottom of the hot leg. This indicator is valved in during the RCS draindown process prior to losing indication on LI-106, the pressurizer wide range level indicator. The reference leg to LI-197 is connected to the bottom of the pressurizer and should provide reasonable accurate level should the RCS become pressurized. Local level indication is provided by a tygon tube connected to the same location on the RCS hot leg as LI-197, and is run up the outside of the biological shield wall and vented to the containment atmosphere near the top of the pressurizer. Because the reference leg is at containment pressure, this local level indication would read high should the RCS become pressurized. Since both of the above level indicators are connected to the bottom of the hot leg that is on the opposite side as the shutdown cooling suction inlet, the indicated levels are not affected by shutdown cooling flow or possible vortexing.

RCS pressure indication is provided by three wide range pressurizer pressure indicators. PI-118 has a range of 0 to 1600 psia and is graduated in increments of 50 psia. UR-113/115 and UR-105/123 are chart recorders both with a low range of 0 to 1000 psia and a high range of 0 to 2500 psia. The strip charts are graduated in increments of 20 psia for the low ranges. All of these instruments are capable of measuring RCS pressure upon loss of the decay heat removal function.

RCS temperature is normally monitored via TR-346, the shutdown cooling inlet and outlet temperature recorder, when the shutdown cooling system is in service. During outages in which the reactor head is not going to be removed, the Core Exit Thermocouples (CETs) would be available to monitor RCS temperature. However, during a refueling outage the CETs are typically disconnected electrically early in the outage in preparation to remove the reactor head. Upon a loss of shutdown cooling, the CETs would be available for RCS temperature monitoring if still connected. Without direct temperature monitoring capabilities, the RCS temperature could be estimated based upon the Heatup Rate Due to Decay Heat versus Time after shutdown curve, found in the Fort Calhoun Technical Data Book, for the appropriate RCS level.

The alarms available to the operators are as follows: LPSI Pump Off Normal - indicates that a running LPSI pump has tripped. SI Pumps Valves Off Normal - alarms when one or more of the suction or discharge valves associated with an SI pump is closed. This alarm, however, annunciates when the shutdown cooling valve line-up is performed. Shutdown Cooling Valves Off Normal--alarms when either of the containment isolation valves (HCV-347 & HCV-348) in the SDC inlet line is closed; or when the shutdown cooling temperature control valve (HCV-341) is closed; or when the shutdown cooling heat exchanger bypass valve (FCV-326) is fully open.

Other indications available include pump amperage for the Safety Injection System Pumps, the shutdown cooling flow rate, and the shutdown cooling heat exchanger outlet temperatures.

The instrumentation identified above, except for the tygon tubing, is permanently installed. The tygon tube is valved out and left in place during plant operation. It is inspected prior to placing in service by operations. The Instrument & Control group performs a calibration on the above listed instrumentation according to the procedural requirements for each instrument.

Item 3

Identification of all pumps that can be used to control NSSS inventory. Include: (a) pumps you require be operable or capable of operation (include information about such pumps that may be temporarily removed from service for testing or maintenance); (b) other pumps not included in item a (above); and (c) an evaluation of items a and b (above) with respect to applicable TS requirements.

Response

Fort Calhoun has two LPSI pumps that are normally used plus three containment spray pumps that can be aligned to provide the residual heat removal service. Both the LPSI pumps and containment spray pumps have a design flow rate of 1500 GPM, the normal shutdown cooling flow setting. Technical Specifications require two decay heat removal loops be operable with one loop in operation when the RCS temperature is between 210°F and 300°F. The steam generators with an associated reactor coolant pump are also considered decay heat removal loops in this temperature range. With RCS temperature below 210°F the Technical Specifications require one decay heat removal loop to be in operation. Additionally, when there is less than 15 feet of water above the core, two decay heat removal loops are required to be operable with one in operation. Fort Calhoun would then have two of the five LPSI or containment spray pumps and both shutdown cooling heat exchangers required to be operable.

Also available to assist in controlling RCS inventory are one of the three HPSI pumps or up to three charging pumps. Because of low temperature overpressure considerations when the reactor vessel head is installed, only one HPSI pump can be enabled below an RCS temperature of 271°F provided the three charging pumps are disabled.

Item 4

A description of the containment closure condition you require for the conduct of operations while the RCS is partially filled. Examples of areas of consideration are the equipment hatch, personnel hatches, containment purge valves, SG secondary-side condition upstream of the isolation valves (including the valves), piping penetrations, and electrical penetrations.

Response

Containment closure requirements are stipulated in the Fort Calhoun Station Technical Specifications. There are no Technical Specification requirements for containment closure that are dependent upon RCS level. The existing requirements, however, are identified below:

Technical Specification 2.6 requires containment integrity be established unless the reactor coolant system is in a cold shutdown condition (less than 210°F) in order to preclude the possibility of steam formation and, hence, any pressure buildup in the containment building. Containment integrity is defined in the Technical Specifications as (1) all nonautomatic containment isolation valves which are not required to be open during accident conditions and blind flanges are closed; (2) the equipment hatch is properly closed; (3) at least one door in the personnel air lock is properly closed; (4) all automatic containment isolation valves are operable or locked closed (or isolated by locked closed valves or blind flanges as permitted by a limiting condition for operation); (5) the uncontrolled containment leakage satisfies Technical Specification 3.5 (specification for containment testing).

Technical Specification 2.8 stipulates a modified form of containment integrity for refueling operations. These requirements are that the equipment hatch and one door in the air lock be properly closed and all automatic containment isolation valves be operable or at least one valve in each line closed.

Technical Specification 2.9 requires that the release of radioactive gaseous effluents be controlled such that the concentration values specified in 10 CFR Part 20, Appendix B are not exceeded in unrestricted areas. In order to ensure compliance with this specification, the Fort Calhoun maintenance procedure for removing the equipment hatch requires one of two conditions to be met prior to removal. The first condition is that the containment atmospheric radionuclide concentration is below a MPCU (maximum permissible concentration unrestricted) fraction of 1. The second condition is that the containment atmospheric radionuclide concentration is below MPCR (maximum permissible concentration restricted) and (1) a barrier preventing air flow or diffusion out of containment is in place across the equipment hatch opening, (2) at least one containment purge fan is taking suction from containment, and (3) the personnel air lock doors are open. Technical Specification 2.9 is often the most limiting and usually precludes opening of the equipment hatch early in an outage.

When in the refueling shutdown condition, Technical Specification 2.1 allows for the decay heat removal loops to be inoperable for up to eight hours provided (1) no operations that could dilute the RCS boron concentration are permitted, (2) no refueling operations are taking place, and (3) all containment penetrations providing direct access from containment atmosphere to the outside atmosphere are closed within four hours.

Item 5

Reference to and a summary description of procedures in the control room of your plant which describe operation while the RCS is partially filled. Your response should include the analytic basis you used for procedures development. We are particularly interested in your treatment of draindown to the condition where the RCS is partially filled, treatment of minor variations from expected behavior such as caused by air entrainment and de-entrainment, treatment of boiling in the core with and without RCS pressure boundary integrity, calculations of approximate time from loss of RHR to core damage, level differences in the RCS and the effect upon instrumentation indications, treatment of air in the RCS/RHR system, including the impact of air upon NSSS and instrumentation response, and treatment of vortexing at the connection of the RHR suction line(s) to the RCS.

Explain how your analytic basis supports the following as pertaining to your facility: (a) procedural guidance pertinent to timing of operations, required instrumentation, cautions, and critical parameters; (b) operations control and communications requirements regarding operations that may perturb the NSSS, including restrictions upon testing, maintenance, and coordination of operations that could upset the condition of the NSSS; and (c) response to loss of RHR, including regaining control of RCS heat removal, operations involving the NSSS if RHR cannot be restored, control of effluent from the containment if containment was not in an isolated condition at the time of loss of RHR, and operations to provide containment isolation if containment was not isolated at the time of loss of RHR (guidance pertinent to timing of operations, cautions and warnings, critical parameters, and notifications is to be clearly described).

Response

OPPD's response to Item 5 will be submitted as discussed in the cover letter.

Item 6

A brief description of training provided to operators and other affected personnel that is specific to the issue of operation while the RCS is partially filled. We are particularly interested in such areas as maintenance personnel training regarding avoidance of perturbing the NSSS and response to loss of decay heat removal while the RCS is partially filled.

Response

The plant operators receive initial and periodic continuing training on plant systems and their operation, including system transients and plant procedures. This training covers plant evolutions including operation while the RCS is partially filled and loss of decay heat removal transients.

Maintenance personnel have received training upon the necessity and importance of procedural compliance. Administrative procedure A-0-20 explicitly places responsibility for the proper tag out of plant equipment with the plant operators. Maintenance or Engineering personnel are not authorized to operate plant equipment that is under operations control unless specifically allowed to do so by the control room operators or shift supervisor. This not only includes the isolation of components or systems, but also the draining of these components or systems.

Item 7

Identification of additional resources provided to the operators while the RCS is partially filled, such as assignment of additional personnel with specialized knowledge involving the phenomena and instrumentation.

Response

Senior licensed operators at Fort Calhoun have received training and are experienced in operating the station while the RCS is at mid-loop. The Company does not currently assign additional personnel to the control room beyond the required control room staffing for this mode of operation. Additional information is available to the operators in the Operation Manual Technical Data Book. Operators utilize this information which presents the relationship between the RCS level indicators LI-106 (the pressurizer wide range level indicator), LI-197 (the refueling level indicator) and the tygon tube versus actual containment elevations. Also, the Technical Data Book displays two graphs, the first plotting decay heat versus time after shutdown and the second plotting the heat-up rate due to decay heat for three common RCS levels (with the centerline of the hot leg included) versus time after shutdown.

Item 8

Comparison of the requirements implemented while the RCS is partially filled and requirements used in other Mode 5 operations. Some requirements and procedures followed while the RCS is partially filled may not appear in the other modes. An example of such differences is operation with a reduced RHR flow rate to minimize the likelihood of vortexing and air ingestion.

Response

Fort Calhoun currently implements no additional requirements while operating with the RCS partially filled. Although not procedurally required, operations normally monitors RCS level by local observations of the tygon tube and compares this observation to the control room level indicator, LI-197, when operating with the RCS at mid-loop. Procedural requirements prohibit draining the RCS below mid-loop. Additionally, the operations staff has the authority to refuse to release any component or system for any activity that has the potential to perturb the NSSS or compromise safe operation of the plant. Fort Calhoun has been able to operate safely with the RCS level at mid-loop and has never lost residual heat removal capabilities due to low RCS level or air entrainment from vortexing.

Item 9

As a result of your consideration of these issues, you may have made changes to your current program related to these issues. If such changes have strengthened your ability to operate safely during a partially filled situation, describe those changes and tell when they were made or are scheduled to be made.

Response

OPPD's response to Item 9 will be submitted with our response to Item 5 as discussed in the cover letter.