Nebraska Public Power District

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U.S. Nuclear Regulatory Commission
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Subject: Reactor Vessel Water Level Instrumentation in BWRs, Generic Letter 84-23, Cooper Nuclear Station NRC Docket No. 50-298, DPR-46

References:
1) Generic Letter 84-23, dated October 26, 1984, "Reactor Vessel Water Level Instrumentation in BWRs"
2) Letter NLS8500119, dated May 31, 1985, from J. M. Pilant (NPPD) to D. B. Vassallo (NRC), Same Subject
3) Letter from D. B. Vassallo (NRC) to J. M. Pilant (NPPD), dated August 21, 1985, "NUREG-0737, Item II.F.2, Inadequate Core Cooling Instrumentation - Safety Evaluation"

Gentlemen:

Generic Letter 84-23 required that the Nebraska Public Power District make appropriate modifications to reactor vessel water level instrumentation to reduce indication errors. The indication errors of concern are those that could be caused by high drywell temperatures. Accordingly, the District submitted the response, listed as Reference 2 above, which identified two possible modifications and a schedule to address the water level instrumentation concerns.

The staff issued a Safety Evaluation Report (Reference 3) stating that either of the two proposed modifications would address the concerns of Generic Letter 84-23. The District decided to install a system to inject core spray make-up water into the reference legs (Option 2 endorsed by the SER). This installation was completed during the 1988 refueling outage.

During post-installation testing, an anomoly was identified in the pressure retaining capability of a solenoid valve, which alters the system design slightly from the design described to the NRC in the letter of May 31, 1985 (Reference 2). The solenoid valve was intended to fulfill two functions: 1) to act along with two check valves as an isolation between the higher pressure reference leg piping and the lower pressure core spray piping and 2) to operate as a remote, manually initiated valve to allow the operator to initiate reference leg make-up flow from the control room. The subject

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solenoid valve, however, will not hold pressure in the reverse flow direction. Accordingly, the solenoid valve is not capable of providing the intended isolation function noted in 1) above. As a result, to ensure that integrity of the reference leg is maintained, a manual isolation valve will be used to isolate the system. System operation, during the current fuel cycle, will require local operation of the manual valve to initiate the make-up flow to the reference legs. Thus, the system is not presently a remote manually operated system, as described to the NRC in response to GL 84-23. Further details are provided in Attachment A.

The Safety Evaluation Report (Reference 3) approves the use of a manually initiated system. The only difference is that, in the interim, manual actuation cannot be initiated from the control room as was described in the District's response. Instead, an operator must open isolation valves, which are located in areas of the plant which would be accessible and radiologically acceptable during the events which might require operation of this system. These isolation valves are located on the 931'-6" elevation of the reactor building with one near instrument rack 25-5 and one near instrument rack 25-6. These locations are shielded from containment by shield walls and would not result in a significant radiological hazard to the operator to access these areas during the events which require operation of this system. Also, the slow progressing nature of the small break LOCA and loss of drywell cooling events are such that operators have sufficient time to open the manual valves. Based upon the above, the District does not believe that the system function or the Safety Evaluation have been greatly affected due to the change from manual system initiation from the control room to manual valve operation locally.

The District fully intends to upgrade the system to allow remote manual operation of the core spray reference leg fill system as originally intended. This upgrade will be completed during the 1989 refueling outage.

In addition to the change described above, several other minor portions of the core spray reference leg fill system are different than the conceptual system design provided in response to GL 84-23. These minor changes do not affect the function of the system, and therefore, do not affect the Safety Evaluation Report prepared by the NRC. A brief description of the changes in the system is provided in Attachment A. A flow diagram (FD-001, Rev. 1) of the Reference Leg Injection System is provided as Attachment B.

The District believes that the changes to the Core Spray Reference Leg Fill System do not invalidate the Safety Evaluation Report previously issued. The changes made do not alter the function of the system nor do they reduce the capability of the system to fulfill its intended function during a small break LOCA or loss of drywell cooling event. With the exception of the remote manual solenoid valve, the changes are the result of detailed design efforts, and reflect considerations which were not fully investigated during the conceptual design stage. The District intends to leave the system as installed except for the remotely operated solenoid valve. Although the District considers the operation of the local manual isolation valve to be an acceptable means of manual initiation, the District intends to upgrade the system to remote operation from the control room for the convenience of the operators. This upgrade will be completed during the 1989 outage.

Should you have any questions or concerns about this issue, please contact this office.

Sincerely,

George A. Trevors
Division Manager
Nuclear Support

GAT/mb:jm31/8(7B)
Attachment

cc: U. S. Nuclear Regulatory Commission
Region IV
Arlington, TX

NRC Senior Resident Inspector
Cooper Nuclear Station

Changes to Core Spray Reference Leg Fill System

1. Remote Manual Operation to Local Manual Operation

The solenoid valves NBI-SOV-SSV-738 and 739, purchased to provide remote manual initiation of the make-up flow to the reference legs, were also intended to provide part of the isolation barrier to prevent flow from the reference legs to the low pressure core spray system. The valve specifications appeared to indicate that this valve, when closed, would retain 1375 psi in either direction. However, during the post installation hydrostatic test, it was discovered that the valve would not hold pressure in the reverse-flow direction. The vendor verified that the valve was only designed and capable of isolation in the forward-flow direction.

Although the hard seated check valves provide an acceptable barrier for piping design pressure change, they will exhibit some leakage, estimated to be less than 3 cc/min. in the reverse flow direction. The District feels it is prudent, and the original implemented design concept intended, to have an additional barrier to prevent any flow of water from the reference leg. To maintain this important design consideration, and as an interim measure until acceptable solenoid valves can be procured and installed, a manual isolation valve, NBI-V-577A(B) located downstream of the two check valves, will be closed to ensure the integrity of the reactor vessel water level reference legs.

The operating procedures have been modified to maintain the manual shutoff valves (NBI-V-577A and B) normally closed in each loop of the core spray reference leg fill system. The operating procedures also provide instructions to the operators to open the manual valves locally, if injection is required. This change in system operation is only intended for one operating cycle, to allow solenoid valves that will isolate in either direction to be obtained and installed during the next refueling outage.

This change does not alter the design of the system. The system was designed to be a manually operated system. With this interim operational modification, remote manual operation will not be possible. Due to the relatively slow progression of events during which this system is required to operate, local manual operation instead of remote manual operation does not functionally change the system design concept. Also, the areas where the manual isolation valves are located (931'-6" elevation of the reactor building, one near instrument rack 25-5 and one near rack 25-6) are accessible and radiologically safe during the small break LOCA and loss of drywell cooling events. The existing Safety Evaluation Report approves the use of a manually initiated system. Therefore, the use of a remote versus a local manually operated system does not invalidate the safety evaluation.

This local manual operation is only intended to be used for one operating cycle. The system will be upgraded during the 1989 outage.

2. Water Filter

The conceptual design, provided to the NRC in Reference 2, suggested installing a filter at the origin of the reference leg piping to assure that the injection water is free of particulate matter. The installed Design Change does not include a filter for the following reasons:

- Discussions with chemistry personnel at CNS revealed that no torus water sample had ever been taken after relief valve discharge into the torus. As a result, the size of particulates which would need to be filtered can not be determined. Typical torus water samples taken on a routine basis did not reveal particulates that were apparent to the naked eye. Inspections of the drained torus revealed that a very fine silt-like material exists at the bottom of the torus. Removal of particulate materials such as these would require a very fine mesh filter.

- If a fine mesh filter were to be installed, it could become plugged with the particulate matter (which might have otherwise passed through the system) and block injection flow. Cleaning/replacement of the filter would require taking the system out of service.

- The system described by the Design Change accommodates the potential for system plugging due to particulate matter, in a manner which is more ALARA efficient than the use of a filter. If particulate matter does plug the limiting component (the orifice in the needle valve, which would be indicated in the control room by inconsistent water level indications), the downstream isolation valve could be manually closed, the drain line valves manually opened, and the needle valve manually opened to 100% open to flush the system and the particulate matter into a nearby radioactive waste drain. This operation can be performed very quickly, relative to filter cleaning/changeout.

3. Restriction Orifice

The conceptual design utilized a restriction orifice to limit the flow to the required rate and to work in conjunction with a differential pressure indicator to measure the flow. The installed design utilizes a needle valve so that the flow can be adjusted in the field to the desired rate. This eliminates the need to depend on a calculation to size the orifice, and gives CNS the flexibility to adjust the flow to any desired rate within the capability of the needle valve.

4. Differential Pressure Indicator

The differential pressure indicator was to be used to locally monitor flow rate during routine surveillance testing. The installed design deleted this indicator when the restriction orifice was deleted. Initial acceptance testing performed to set the needle valve position, utilized direct measurement of flow from the drain connection to verify equivalent flow to the reference leg of 0.2 to 0.5 gpm.

5. Differential Pressure Switch

A differential pressure switch was included in the conceptual design to prevent the solenoid valve from opening if the line pressure downstream of the solenoid valve (reference leg pressure) is greater than the line pressure upstream of the solenoid valve (core spray pressure), thus preventing overpressurization of the core spray system. However, the two check valves in series perform this same function, and existing core spray system relief valves 11-RV-17 (Loop A) and 13-RV-17 (Loop B) will protect the piping if leakage past the check valves and the solenoid valve result in pressure in the core spray piping greater than design. In addition, a high pressure condition in the core spray piping would be alarmed in the control room.

6. ASME Sec. III Class 1 Code Designation

The conceptual design identified two alternatives for providing core spray make-up to the reference legs. One alternative was to inject water into the ASME Class 1 piping inside the drywell. The other alternative allowed injecting the core spray water at the reactor water level instrument racks. This second alternative was chosen. Since the make-up water is not injected into the Class 1 portion of the reference leg piping, but instead, is injected downstream of the Class 1 piping boundary defined by the excess flow check valve, the injection piping is designated as Class 2.

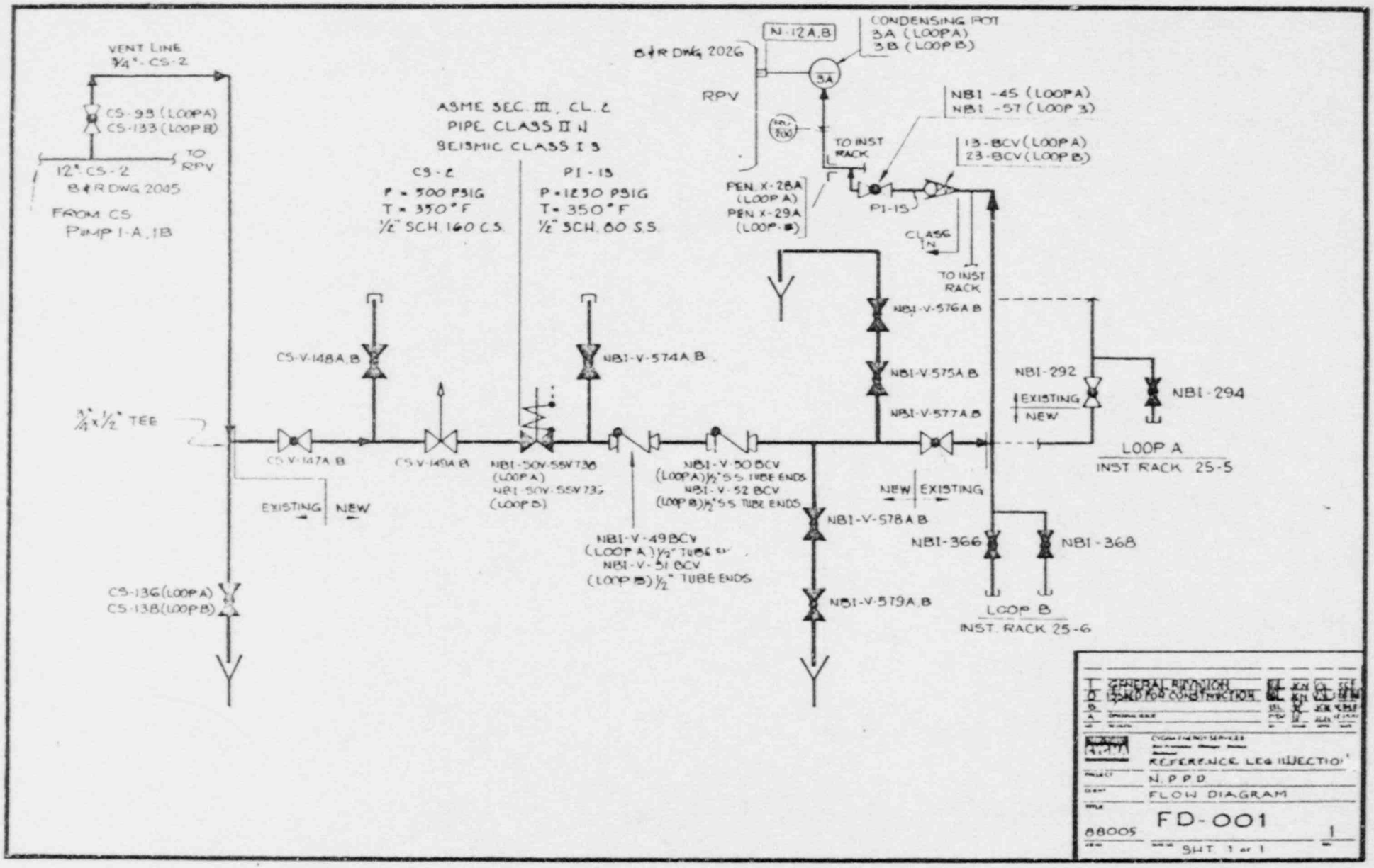
7. Reference Leg Insulation Inside Containment

The conceptual design called for insulation on the reference leg piping inside containment to avoid flashing or boil-off due to high drywell temperature. As a part of Design Change 87-113, the District performed heat transfer calculations under postulated temperature and pressure conditions where reference leg injection is required. These calculations demonstrate that insulation is not required to ensure injected water does not reach saturation prior to arriving at the condensing pots. Therefore, no insulation was installed on reference leg piping.

8. Error Due to Injection Flow

The conceptual design, which was developed by General Electric and submitted to the NRC in response to Generic Letter 84-23, stated that the maximum error induced in the indicated water level is 2 inches of water. This induced error is caused by the flow restriction of the 1/4 inch restricting orifice in the reference legs and an injection flow rate of 0.5 gpm. Detailed calculations performed as a part of Design Change 87-113 show that the actual maximum error induced at 0.5 gpm injection flow rate is 7 inches of water. This discrepancy in the conceptual design was due to the assumption that the existing cold reference leg restriction devise was a smooth edged venturi. The actual restricting orifice installed at CNS is a square edged device, resulting in the higher pressure drop.

This is not considered a significant concern since the error will be induced in the conservative direction. That is, the induced error will show reactor water level to be lower than actual reactor vessel water level. This small error, therefore, could only lead to operator actions or trip functions or ECCS initiation taking place at a water level that is actually 7 inches higher than indicated. The result is that action is taken while reactor water level is at a slightly more conservative (higher than indicated) water level. Based upon the above, the small error induced by the reference leg injection system does not pose a safety concern and the NRC Safety Evaluation, which found the error to be acceptable, is still valid.



ATTACHMENT B