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SUBJECT: Forwards response to NRC 890210 ltr re design basis & criteria used for reactor bldg closed cooling water sys.

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July 14, 1989

10 CFR Part 50
Appendix A
TAC No. 71866

Director of Nuclear Reactor Regulation
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MONTICELLO NUCLEAR GENERATING PLANT
DOCKET NO. 50-263 LICENSE NO. DPR-22

Design Basis and Criteria for the
Reactor Building Closed Cooling Water System and
its Isolation Provisions

References: (1) Letter from J J Stefano (NRC), to D M Musolf (NSP), "Reactor Building Closed Cooling Water (RBCCW) System at the Monticello Nuclear Generating Plant (TAC NO. 71866)", dated February 10, 1989.

Attachment 1 to this letter provides the information requested by the NRC Staff in Reference (1). It provides the design basis and criteria used for the Reactor Building Closed Cooling Water (RBCCW) System and its isolation provisions.

Please contact us if you have any questions or further information is necessary related to this issue.

Thomas M Parker
Manager
Nuclear Support Services

c: Regional Administrator - III, NRC
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Attachment

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MONTICELLO NUCLEAR GENERATING PLANT

Attachment 1

INTRODUCTION

This description provides the original design basis for the RBCCW system and its isolation provisions, discussion of the control room indications and procedures currently used to determine when to isolate RBCCW, a discussion on the extent to which the provisions of NUREG-0737 Item II.E.4.2 are being met for the RBCCW System, and a discussion of the acceptability of our isolation provisions for the RBCCW System.

I. DESIGN BASES

The design bases for the RBCCW System isolation provisions were established well in advance of the guidance provided in Section 6.2.4 of the Standard Review Plan and the General Design Criteria of Appendix A to 10 CFR 50. The only guidance at the time was the 1967 AEC Proposed Rule Making which introduced Appendix A. Guidance on containment isolation was provided in Criterion 53, which states "Penetrations that require closure for the containment function shall be protected by redundant valving and associated apparatus." General Electric's Design Specifications for the primary containment system and the containment isolation system were more specific.

With respect to the RBCCW System isolation provisions the following General Electric Specifications define the original design bases. Section 4.3.2.4 of Specification 22A1126, Primary and Secondary Reactor Containment System, Rev. 2, Sept. 30, 1969, states "Lines which do not open into the drywell, and whose external branches do not terminate in dead end service capable of withstanding drywell design conditions, shall utilize one remotely operable isolation valve or one check valve (Example: Closed cooling water lines)." Section 3.2.2 of Specification 22A1132, Containment Isolation Systems, Rev. 3, April 26, 1971, states "Class C valves are on process lines that penetrate the primary containment but do not communicate directly with the reactor vessel or with the primary containment free space and are not on lines that communicate with the environs. These lines require one valve, outside the primary containment, that closes automatically by process action (reverse flow,etc.) or by remote manual operation." In Appendix A to this specification RBCCW isolation valves are identified as Class C. These design bases were followed and are discussed in our USAR.

Section 5.2.1.2.2 of the USAR describes the general criteria governing isolation valves. For closed systems inside containment the following is stated " Lines which penetrate the primary containment and neither connect to the reactor primary system nor open into the primary containment, are provided with at least one valve that is located outside the primary containment. Valves in this category are capable of remote actuation from

the main control room." Specific reference to RBCCW is provided in Section 5.2.2.5.3 which states "Lines, such as those of the reactor building closed cooling water system lines which do not connect to the reactor primary system or open into the primary containment, are provided with at least one a-c power valve located outside the primary containment or a check valve on the influent line inside the containment." Table 5.2-3a identifies the containment isolation valves for penetration X-23 as RBCC-15 and MO-4229 and for penetration X-24 as MO-1426 and MO-4230. MO-4229 and MO-4230 were added in 1987 in conjunction with modifications to make both penetrations testable for 10CFR50 Appendix J.

The RBCCW System was not designed as a safety related system. Therefore, it did not meet AEC Criteria 40 and 42 for missile protection and effects of a loss-of-coolant accident for engineered safety features. However, this non-safety related system was expected to provide some essential cooling. General Electric Specification 22A1211, Rev 1, Oct 12, 1967 states, "Cooling shall be maintained on certain critical equipment during failure of off-site AC power. At such times the non-essential equipment shall be shutoff by closing Valve A on Figure I and the essential Equipment indicated on Figure I (RHR Pumps, Drywell Coolers, Recirc Pumps and Drywell Sump) supplied with cooling water of the amounts required by Table II. Electric power for operating the cooling water system during such periods shall be supplied by the diesel generator." The USAR also identifies that the system is not safety related. Section 10.4.3.3 states "The reactor building closed cooling water system is not required during or immediately subsequent to a design basis accident and is therefore not safety related."

In summary, the Monticello RBCCW System isolation provisions are in conformance with the original design bases and original licensing criteria.

II. CONTROL ROOM INDICATION AND ISOLATION PROCEDURES

Indication of RBCCW leakage/line break inside primary containment is provided by several different alarms in the control room. The indications can be divided into those for which manual isolation would be initiated during normal operation and those for which manual isolation would be initiated following a LOCA.

During normal operation the following control room alarms would provide indication of possible RBCCW leakage/line break inside containment:

ALARM	DESCRIPTION/ACTION CONDITIONS
4-B-5	Recirc Pump A Low Cooling Water Flow/ RBCCW system pressure less than 30 psig as indicated by PI-1399 on C06

4-B-10	Recirc Pump B Low Cooling Water Flow/ RBCCW system pressure less than 30 psig as indicated by PI-1399 on C06
4-B-11	Reactor Water Cleanup Pump Water Temp High/ RBCCW system pressure less than 30 psig as indicated by PI-1399 on C06
4-B-17	Drywell Floor Drain Sump High Level/ Increased drywell sump levels/flows as indicated by LR-7409 on C03/FR-2544 on C04 coincident with alarms 6-B-27 and/or 6-B-32
4-B-23	Drywell Floor Drain Sump Leak Rate High/ Increased drywell sump levels/flows as indicated by LR-7409 on C03/FR-2544 on C04 coincident with alarms 6-B-27 and/or 6-B-32
4-B-28	Drywell Floor Drain Sump Leak Rate Change High/ Increased drywell sump levels/flows as indicated by LR-7409 on C03/FR-2544 on C04 coincident with alarms 6-B-27 and/or 6-B-32
6-B-27	RBCCW Surge Tank T-3 High-Low Level/ Coincident with increased drywell sump levels/flows as indicated by LR-7409 on C03 /FR-2544 on C04
6-B-32	RBCCW Low Discharge Pressure/ RBCCW system pressure less than 30 psig as indicated by PI-1399 on C06
6-B-33	RBCCW Standby Pump Start/ RBCCW system pressure less than 30 psig as indicated by PI-1399 on C06

When one of these alarms is received the control room operator is directed to the Abnormal Operating Procedures if the coincident conditions associated with the alarm identified above are met. Abnormal Procedure C.4-B.4.1.F, Leak Inside Primary Containment, would be entered if drywell floor drain sump instrumentation indicated an increase in leakage. Abnormal Procedure C.4-B.2.5.A, Loss of RBCCW Flow, would be entered if RBCCW system pressure was less than 30 psig. This procedure will attempt to restart RBCCW pumps to re-establish system pressure and if system pressure can not be restored, will trip the Recirc. pumps and isolate RBCCW containment isolation valves.

Following a LOCA, control room alarms would provide indication of possible RBCCW leakage/line break inside containment:

ALARM	DESCRIPTION/ACTION CONDITIONS
4-B-5	Recirc Pump A Low Cooling Water Flow/ RBCCW system pressure less than 30 psig as indicated by PI-1399 on C06
4-B-10	Recirc Pump B Low Cooling Water Flow/ RBCCW system pressure less than 30 psig as indicated by PI-1399 on C06
4-B-11	Reactor Water Cleanup Pump Water Temp High/ RBCCW system pressure less than 30 psig as indicated by PI-1399 on C06
6-B-32	RBCCW Low Discharge Pressure/ RBCCW system pressure less than 30 psig as indicated by PI-1399 on C06
6-B-33	RBCCW Standby Pump Start/ RBCCW system pressure less than 30 psig as indicated by PI-1399 on C06

When one of these alarms is received, the control room operator is directed to the Abnormal Operating Procedures if the coincident conditions associated with the alarm identified above are met. Abnormal Procedure C.4-B.2.5.A, Loss of RBCCW Flow, would be entered if RBCCW system pressure was less than 30 psig. Entry to this procedure following a LOCA would occur if there was a loss of offsite power or a break of one of the main RBCCW headers in the drywell. The operators are instructed to manually isolate RBCCW even though no break or leak may exist.

If for some reason RBCCW was not isolated when any of the above alarms were received following a LOCA, the following alarms could possibly annunicated if a leakage path existed from primary containment out through the RBCCW surge tank vent and overflow lines:

ALARM	DESCRIPTION/ACTION CONDITIONS
4-A-6	New Fuel Storage Area High Radiation/ ARM A-4 Trip on Reactor Building Elev. 1001
4-A-11	Reactor Building High Radiation/ Various ARM Trips
SPDS	Area Radiation Monitors in alarm

When one of these alarms is received the control room operator is directed to the Emergency Operating Procedures. EOP C.5-1302, Secondary Containment Radiation Control, would be entered. If ARM-4 or ARM-7 are reading off-scale, the operator is directed to isolate all systems that are discharging into the area. RBCCW is a system that could be discharging into the areas associated with these ARM's.

In summary, Monticello has abundant and diverse indication and procedures to provide an adequate and acceptable remote isolation function.

III. BREAK SIZE

During normal operation, 5 gpm is the smallest leakage rate (break size) in which manual isolation of RBCCW would be considered. After a leak inside the drywell is identified by one of the previously identified alarms, Abnormal Procedure C.4-B.4.1.F, Leak Inside Primary Containment, would be entered. This procedure requires that "IF the floor drain leak rate is greater than 5 gpm, THEN reduce the leak rate to less than 5 gpm within 4 hours or initiate an orderly Reactor shutdown and reduce Reactor water temperature to less than 212°F within 24 hours." If RBCCW is identified as the source of the leak, instructions are given to consider isolating RBCCW.

During a LOCA, the size of the break in which manual isolation would result is inconsequential. Any RBCCW break inside the drywell will eventually cause RBCCW system pressure to drop to 30 psig. At 30 psig system pressure, Abnormal Procedure C.4-B.2.5.A, Loss of RBCCW Flow, would be entered. This procedure requires manual isolation of RBCCW containment isolation valves if system pressure can not be restored.

IV. PROBABILISTIC RISK ASSESSMENT

A study was performed to compare the risks of the current RBCCW isolation provisions with the risks of automatic isolation of RBCCW.

The probability of core damage LOCA events which do not result in containment failure was multiplied by an estimated probability of a LOCA causing RBCCW line failure in the drywell and an estimated probability of the operator failing to isolate the line as directed by procedures. For RBCCW with automatic isolation, the probability of isolation valve failure was substituted for operator failure.

The probability of a LOCA causing a RBCCW line break which results in a release to the secondary containment is 7.5E-8/year without automatic isolation and 9.4E-10 with automatic isolation.

This result was based on the following factors and considerations:

1. The core damage frequency from all LOCA events which do not result in containment failure is about 1.5E-6/year (class IIIA + IIIB + IIIC) based on the results of the Monticello IPE which is based on the IDCOR IPE Methodology.

2. External events are not considered.

3. A probability of 0.5 was used to estimate the probability of any LOCA breaking the RBCCW line. This is judged to be conservative because of the potential for observing a leak before break and because many breaks would most likely not cause a break of the RBCCW line because the break is not in the vicinity of the RBCCW piping or that the break is too small.

4. A probability of 0.1 was used for the operator failing to isolate the break assuming a high stress, non routine action in a 30 minute time frame, based on WASH 1400.

5. A probability of 1.25E-3 was used for isolation valve failing to operate, based on WASH 1400.

6. The leakage out of the RBCCW surge tank vent line is assumed to not fail any vital ECCS equipment because of the small size of the line and the location of the leakage on the third or fourth floors while the ECCS equipment is in lower areas of the building.

7. The assessment of the automatic isolation function did not take into account the increased risk due to inadvertent automatic isolations that could adversely impact safety (i.e., drywell temperature and pressure increases, and recirc pump seal failures).

The following conclusions can be made from this assessment:

1. Failing to manually isolate RBCCW when required to maintain primary containment integrity is a very low probability event.

2. A modification to automatically isolate RBCCW containment valves would not result in a meaningful reduction in risk.

3. A detailed cost benefit analysis, using a risk assessment improvement as the benefit, could not justify the cost for a modification to install automatic isolation provisions.

V. NUREG-0737 ITEM II.E.4.2 COMPLIANCE

Position 2 and position 3 of NUREG-0737 Item II.E.4.2 are the main issues with respect to the RBCCW isolation provisions. Position 2 states "All plant personnel shall give careful consideration to the definition of

essential and nonessential systems, identify each system determined to be essential, identify each system determined to be nonessential, describe the basis for selection of each essential system, modify their containment isolation designs accordingly, and report the results of the re-evaluation to the NRC." Position 3 states "All nonessential systems shall be automatically isolated by the containment isolation signal." In our response to the recommendations of NUREG-0578 Item 2.1.4, which is identical to NUREG-0737 Item II.E.4.2 positions 1 thru 4, we defined the criteria used to classify a system as essential. Systems or portions of system are classified as essential if they fulfill one or more of the following functions:

1) Operate to prevent an accident
2) Operate to mitigate the consequences of an accident
3) Operate to assist a system to fulfill its safeguard function
4) Available as a means of diagnosing accident conditions
5) Available to mitigate the consequences of an accident

At that time we identified the RBCCW system as nonessential. Also, we identified that the system is a closed system which is not open to the reactor or containment and that a failure of the piping would have to occur to breach containment. From our response the NRC concluded that "all systems classified as non-essential are isolated by automatic diverse signals to an acceptable degree." The NRC's acceptance of our response to NUREG-0578 came in March, 1980. NUREG-0737 was issued in October, 1980. Since our response to Recommendations 1 thru 4 of NUREG-0578, which are identical to Positions 1 thru 4 of Item II.E.4.2 of NUREG-0737, had already been accepted, no further reviews were done when NUREG-0737 was issued. Item 10 of NRC Inspection Report, dated December 22, 1981, identified that there were no open issues with respect to NUREG-0737 Item II.E.4.2 Positions 1 thru 4. This was taken as confirmation of the acceptability of our earlier response to these positions.

The issue of RBCCW isolation provisions has been reviewed by the BWR Owners Group in NEDC-22253. The conclusions reached for operating plants were:

> Based on information provided in a conference call with the NRC on April 1, 1982, it was not the intention of the NRC to backfit the Standard Review Plan (SRP) Section 6.2.4 definition of a "closed system inside containment" via NUREG-0737 Item II to operating plants. Therefore, the RBCW and DWCW system configurations and isolation provisions already licensed are acceptable, provided:
>
> 1) at least one isolation valve is provided on each line as close to the containment as practical,
>
> 2) the isolation valves are automatic or capable of remote manual operation,
>
> 3) power supplies and controls for the isolation valves are safety grade (classified important to safety), and
>
> 4) piping systems from the containment out to, and including, the isolation valves are safety grade.

Monticello's current system configuration and isolation provision meet the criteria set above.

VI. SUMMARY

The Monticello RBCCW System isolation provisions are in conformance with the original design bases and original licensing criteria. Monticello has abundant and diverse indication and procedures to provide an adequate and acceptable remotely isolation function. A Probabilistic Risk Assessment showed that automatic isolation of RBCCW does not significantly reduce risk. Monticello's current system configuration and isolation provision meet the criteria outlined by the BWR Owner's Group. Therefore, the isolation provisions for the RBCCW system are satisfactory in its current configuration.