1. (MP&L P/L Item No. 001)

SUBJECT: Automatic Depressurization System (ADS) Valve Operability Requirements Technical Specification 3.5.1 and Bases 3/4.5.1 and 3/4.5.2.

DESCRIPTION OF CHANGE: Revisions to Technical Specifications 3.5.1.a.3 and 3.5.1.b.2 and Bases 3/4.5.1 and 3/4.5.2 are proposed to achieve consistency between the technical specifications and the plant's design and accident analyses.

1. Specifications 3.5.1.a.3 and 3.5.1.b.2 should be revised to require eight (8) operable ADS valves instead of seven (7). (Page 3/4 5-1.)

2. Bases 3/4.5.1 and 3/4.5.2 should be revised to indicate that although the ADS controls eight selected valves, the safety analysis takes credit for seven of these valves. (Page B 3/4 5-2.)

   In addition, in the same Bases section, an editorial revision is proposed to correct a typographical error. (Page B 3/4 5-1.)

JUSTIFICATION: In the Grand Gulf design, the ADS controls eight safety-relief valves. If the High Pressure Core Spray (HPCS) system fails to function properly after a small break loss-of-coolant accident (LOCA), the ADS automatically causes selected safety-relief valves to open, depressurizing the reactor so that flow from the low pressure core cooling systems can enter the core in time to limit fuel cladding temperature to less than the maximum value allowed by 10 CFR 50.46, i.e., 2200°F.

Significant input parameters for the Grand Gulf LOCA analyses are presented in FSAR Table 6.3-2. This table indicates that eight ADS valves are assumed operable for the LOCA analyses.

The plant's response to the most limiting small break LOCA is discussed in the FSAR, in response to NRC Question 212.24 (as amended, November, 1981). This particular accident analysis series postulated a HPCS line break coincidental with the worst case single active failure, namely, loss of the Division I diesel generator. Under these assumptions all credit for core spray cooling is removed. The maximum peak cladding temperature (PCT) in these analyses was determined to be approximately 1824°F and occurred at a break size of approximately 0.01 ft^2. In that these analyses assumed eight ADS valves to be operable, Technical Specification 3.5.1 should be revised, as discussed above, to be consistent with this assumption, i.e., require eight valves to be operable as a Limiting Condition for Operation. Additional, more restrictive analyses were performed in support of the proposed changes to Bases 3/4.5.1 and 3/4.5.2. These analyses assumed one ADS valve inoperable and are discussed below.

Action Statement e.1 of the subject specification permits one ADS valve to be inoperable for a duration of up to 14 days prior to requiring a plant shutdown. In order to ensure safe operation during this extended period of time with one ADS valve out of service, a series of small break analyses were performed utilizing assumptions identical to those discussed above except that only seven ADS valves were assumed operable. The maximum PCT developed in these analyses was approximately 2064°F, which occurred at a break size of 0.015 ft^2. This maximum PCT meets the acceptance criteria of 10 CFR 50.46 (2200°F) and is also below the maximum PCT determined in Grand Gulf's most limiting LOCA analysis, which considers the entire break spectrum (2098°F, FSAR Table 6.3-3).

Bases 3/4.5.1 and 3/4.5.2 should be revised to indicate that although the ADS controls <u>eight</u> safety-relief valves, the safety analyses support an assumption of <u>seven</u> operable ADS valves, consistent with the above discussion. The FSAR will be revised, specifically Section 6.3.3 and the response to NRC Question 212.24, to reflect the above analyses and results associated with the assumption of one ADS valve out of service.

The change to page B 3/4 5-1 is proposed to correct a typographical error and is purely administrative in nature.

SIGNIFICANT HAZARDS CONSIDERATION:

The proposed changes to the technical specifications have been evaluated to involve no significant hazard, as defined in 10 CFR 50.92. The changes have been proposed to render the technical specifications consistent with the safety analyses and the plant design. The ADS system controls <u>eight</u> safety-relief valves; however, safety analyses have been conducted to demonstrate that plant performance with seven ADS valves operable meets the applicable NRC acceptance criteria, in particular, 10 CFR 50.46.

MP&L considers the proposed changes to be similar to examples of proposed amendments that are not likely to involve a significant hazards consideration. These examples and other guidance were provided by the NRC in the Federal Register, dated April 6, 1983 (48 FR 14870). The specific examples noted are listed as follows:

(ii) A change that constitutes an additional limitation, restriction, or control not presently included in the technical specification; and

(vi) A change which either may result in some increase to the probability or consequences of a previously analyzed accident or may reduce in some way a safety margin, but where the results of the change are clearly within all acceptable criteria with respect to the system or component specified in the Standard Review Plan.

Example (ii) is applicable due to the more restrictive operability requirements imposed by the proposed change to Technical Specification 3.5.1. Example (vi) applies primarily to the analyses performed in support of the proposed changes to Bases 3/4.5.1 and 3/4.5.2. These analyses assumed only seven ADS valves to be operable and demonstrated that the maximum PCT for this series of accidents was clearly within the applicable NRC acceptance criteria.

In that the proposed changes promote consistency between the plant design, the safety analyses, and the technical specifications and in that the supporting safety analyses were carried out in accordance with 10 CFR 50.46 and Appendix K, the proposed changes are not considered to:

1. Involve a significant increase in the probability or consequences of an accident previously evaluated; or

2. Create the possibility of an accident of a type different from any evaluated previously; or

3. Involve a significant reduction in a margin of safety.

Therefore, the proposed changes to the technical specifications were determined to involve no significant hazards considerations.

3/4.5 EMERGENCY CORE COOLING SYSTEMS

3/4.5.1 ECCS - OPERATING

LIMITING CONDITION FOR OPERATION

3.5.1 ECCS divisions 1, 2 and 3 shall be OPERABLE with:

a. ECCS division 1 consisting of:

1. The OPERABLE low pressure core spray (LPCS) system with a flow path capable of taking suction from the suppression pool and transferring the water through the spray sparger to the reactor vessel.

2. The OPERABLE low pressure coolant injection (LPCI) subsystem "A" of the RHR system with a flow path capable of taking suction from the suppression pool and transferring the water to the reactor vessel.

3. ~~At least 7~~ Eight OPERABLE ADS valves.

b. ECCS division 2 consisting of:

1. The OPERABLE low pressure coolant injection (LPCI) subsystems "B" and "C" of the RHR system, each with a flow path capable of taking suction from the suppression pool and transferring the water to the reactor vessel.

2. ~~At least 7~~ Eight OPERABLE ADS valves.

c. ECCS division 3 consisting of the OPERABLE high pressure core spray (HPCS) system with a flow path capable of taking suction from the suppression pool and transferring the water through the spray sparger to the reactor vessel.

APPLICABILITY: OPERATIONAL CONDITION 1, 2*# and 3*.

ACTION:

a. For ECCS division 1, provided that ECCS divisions 2 and 3 are OPERABLE:

1. With the LPCS system inoperable, restore the inoperable LPCS system to OPERABLE status within 7 days.

2. With LPCI subsystem "A" inoperable, restore the inoperable LPCI subsystem "A" to OPERABLE status within 7 days.

3. With the LPCS system inoperable and LPCI subsystem "A" inoperable, restore at least the inoperable LPCI subsystem "A" or the inoperable LPCS system to OPERABLE status within 72 hours.

4. Otherwise, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

*The ADS is not required to be OPERABLE when reactor steam dome pressure is less than or equal to 135 psig.

#See Special Test Exception 3.10.5.

3/4.5.1 and 3/4.5.2 ECCS - OPERATING and SHUTDOWN

ECCS division 1 consists of the low pressure core spray system and low pressure coolant injection subsystem "A" of the RHR system and the automatic depressurization system (ADS) as actuated by trip system "A". ECCS division 2 consists of low pressure coolant injection subsystems "B" and "C" of the RHR system and the automatic depressurization system as actuated by trip system "B".

following

The low pressure core spray (LPCS) system is provided to assure that the core is adequately cooled ~~folowing~~ a loss-of-coolant accident and, together with the LPCI system, provides adequate core cooling capacity for all break sizes up to and including the double-ended reactor recirculation line break, and for smaller breaks following depressurization by the ADS.

The LPCS is a primary source of emergency core cooling after the reactor vessel is depressurized and a source for flooding of the core in case of accidental draining.

The surveillance requirements provide adequate assurance that the LPCS system will be OPERABLE when required. Although all active components are testable and full flow can be demonstrated by recirculation through a test loop during reactor operation, a complete functional test requires reactor shutdown. The pump discharge piping is maintained full to prevent water hammer damage to piping and to start cooling at the earliest moment.

The low pressure coolant injection (LPCI) mode of the RHR system is provided to assure that the core is adequately cooled following a loss-of-coolant accident. The LPCI system, together with the LPCS system, provide adequate core flooding for all break sizes up to and including the double-ended reactor recirculation line break, and for small breaks following depressurization by the ADS.

The surveillance requirements provide adequate assurance that the LPCI system will be OPERABLE when required. Although all active components are testable and full flow can be demonstrated by recirculation through a test loop during reactor operation, a complete functional test requires reactor shutdown. The pump discharge piping is maintained full to prevent water hammer damage to piping and to start cooling at the earliest moment.

ECCS division 3 consists of the high pressure core spray system. The high pressure core spray (HPCS) system is provided to assure that the reactor core is adequately cooled to limit fuel clad temperature in the event of a small break in the reactor coolant system and loss of coolant which does not result in rapid depressurization of the reactor vessel. The HPCS system permits the reactor to be shut down while maintaining sufficient reactor vessel water level inventory until the vessel is depressurized. The HPCS system operates over a range of 1160 psid, differential pressure between reactor vessel and HPCS suction source, to 0 psid.

The capacity of the system is selected to provide the required core cooling. The HPCS pump is designed to deliver greater than or equal to 1440/5010 gpm at differential pressures of 1160/200 psi. Initially, water from the condensate storage tank is used instead of injecting water from the suppression pool

ECCS-OPERATING and SHUTDOWN (Continued)

into the reactor, but no credit is taken in the safety analyses for the condensate storage tank water.

With the HPCS system inoperable, adequate core cooling is assured by the OPERABILITY of the redundant and diversified automatic depressurization system and both the LPCS and LPCI systems. In addition, the reactor core isolation cooling (RCIC) system, a system for which no credit is taken in the safety analysis, will automatically provide makeup at reactor operating pressures on a reactor low water level condition. The HPCS out-of-service period of 14 days is based on the demonstrated OPERABILITY of redundant and diversified low pressure core cooling systems.

The surveillance requirements provide adequate assurance that the HPCS system will be OPERABLE when required. Although all active components are testable and full flow can be demonstrated by recirculation through a test loop during reactor operation, a complete functional test with reactor vessel injection requires reactor shutdown. The pump discharge piping is maintained full to prevent water hammer damage and to provide cooling at the earliest moment.

Upon failure of the HPCS system to function properly after a small break loss-of-coolant accident, the automatic depressurization system (ADS) automatically causes selected safety-relief valves to open, depressurizing the reactor so that flow from the low pressure core cooling systems can enter the core in time to limit fuel cladding temperature to less than 2200°F. ADS is conservatively required to be OPERABLE whenever reactor vessel pressure exceeds 135 psig even though low pressure core cooling systems provide adequate core cooling up to 350 psig.

ADS automatically controls ~~seven~~ eight selected safety-relief valves although the safety analysis only takes credit for ~~six~~ seven valves. It is therefore appropriate to permit one valve to be out-of-service for up to 14 days without materially reducing system reliability.

3/4.5.3 SUPPRESSION POOL

The supression pool is required to be OPERABLE as part of the ECCS to ensure that a sufficient supply of water is available to the HPCS, LPCS and LPCI systems in the event of a LOCA. This limit on suppression pool minimum water volume ensures that sufficient water is available to permit recirculation cooling flow to the core. The OPERABILITY of the suppression pool in OPERATIONAL CONDITIONS 1, 2 or 3 is required by Specification 3.6.3.1.

Repair work might require making the suppression pool inoperable. This specification will permit those repairs to be made and at the same time give assurance that the irradiated fuel has an adequate cooling water supply when the suppression pool must be made inoperable, including draining, in OPERATIONAL CONDITION 4 or 5.

In OPERATIONAL CONDITION 4 and 5 the suppression chamber minimum required water volume is reduced because the reactor coolant is maintained at or below 200°F. Since pressure suppression is not required below 212°F, the minimum required water volume is based on NPSH, recirculation volume, and vortex prevention plus a 1'2" safety margin for conservatism.

2. (MP&L P/L Item No. 005)

SUBJECT: Technical Specification Tables 3.3.2-1 and 4.3.2.1-1 item 4.h, pages 3/4 3-12, 3/4 3-14, 3/4 3-22, and 3/4 3-23a.

DESCRIPTION OF CHANGE: This proposed change to Technical Specification Table 3.3.2-1 item 4.h, consists of three parts:

1. Changing the minimum operable channels per trip system from "NA" to "1" for the SLCS initiation RWCU isolation function;

2. Replacing Operational Condition 3 with Operational Condition 5 and adding footnote "##" which requires the SLCS initiation RWCU isolation function to be operable in Operational Condition 5 only when control rods are withdrawn;

3. Replacing present Action 27 for the SLCS initiation RWCU isolation function with new Action 30 on Table 3.3.2-1, which requires the affected SLCS pump to be declared inoperable whenever the associated SLCS initiation instrumentation is inoperable.

The proposed change described in 2. above should also be provided in Technical Specification Table 4.3.2.1-1, item 4.h.

JUSTIFICATION: In order to enter Action b or c of Technical Specification 3.3.2, the number of operable channels must be less than that required by the Minimum Operable Channels per Trip System in Table 3.3.2-1. Presently no channels are required to be operable for the RWCU isolation function upon SLCS initiation. Therefore, the action statements for Technical Specification 3.3.2 can never be entered for this RWCU isolation function. The SLCS initiation RWCU isolation function design consists of one channel per trip system. Requiring one channel per trip system to be operable will reflect system design and require entering appropriate action statements for inoperable channels.

The present applicable operational conditions for the SLCS in Technical Specification 3/4.1.5 are not the same as the ones for the SLCS initiation RWCU isolation function in Tables 3.3.2-1 and 4.3.2.1-1. Technical Specification 3/4.1.5 requires the SLCS to be operable in Operational Conditions 1, 2, and 5* where the "*" footnote applies with any control rod withdrawn but is not applicable to control rods removed per Technical Specification 3.9.10.1 or 3.9.10.2. The applicable operational conditions for the SLCS initiation isolation function in Tables 3.3.2-1 and 4.3.2.1-1 are 1, 2, and 3. The applicable operational conditions for these specifications should be identical since the specifications involve the same SLC system. Also, since control rods cannot be pulled in Operational Condition 3, the SLCS and the associated RWCU isolation function are not required to be operational. This part of the proposed Technical Specification change deletes

Operational Condition 3 and adds 5## to item 4.h of Tables 3.3.2-1 and 4.3.2.1-1 (the "*" footnote from Technical Specification 3.1.5 and the new "##" footnote are identical).

Present Action 27 for the SLCS initiation RWCU isolation function in Table 3.3.2-1 requires the RWCU isolation valves to be closed and the RWCU system to be declared inoperable whenever the associated SLCS initiation instrumentation is inoperable. This action can have an adverse impact on reactor water quality at power. The new proposed Action 30 for the SLCS initiation function will require that the affected SLCS pump be declared inoperable. This new Action 30 will then require entry into Action a.1 or b.1 of Technical Specification 3.1.5 to determine the appropriate action requirements for an inoperable SLCS pump. There are two SLC systems in the Grand Gulf design. SLCS "A" initiation will close RWCU outboard isolation valve G33-F004. SLCS "B" initiation will close RWCU inboard isolation valves G33-F001 and outboard isolation valve G33-F251. Initiation of either SLCS "A" or "B" will cause isolation of the RWCU system; therefore, one SLCS pump can be declared inoperable without adversely affecting the isolation capability of the RWCU, since the RWCU system will isolate if the remaining SLC system is initiated.

SIGNIFICANT HAZARDS CONSIDERATION:

Requiring one minimum operable channel per trip system for the SLCS initiation isolation function constitutes an additional limitation not presently in the Technical Specifications.

The change to the applicable operational conditions is made to promote consistency among Technical Specification 3.1.5, Table 3.3.2-1, and Table 4.3.2.1-1.

Changing the action statement for the SLCS initiation function from the present Action 27 to the new Action 30 on Table 3.3.2-1 is made to ensure reactor water quality, by not isolating the RWCU, and still retain the isolation function from the redundant SLCS system. Also, since the affected SLCS pump is declared inoperable by the new Action 30, the SLCS initiation function must be restored within the time constraints of the action statements of Technical Specification 3.1.5.

The proposed changes do not:

1. Involve significant increase in the probability or consequences of an accident previously evaluated; or

2. Create the possibility of an accident of a type different from any evaluated previously; or

3. Involve a significant reduction in a margin of safety.

Therefore, the proposed changes do not involve a significant hazards consideration.

TABLE 3.3.2-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION

TRIP FUNCTION	VALVE GROUPS OPERATED BY SIGNAL (a)	MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM (b)	APPLICABLE OPERATIONAL CONDITION	ACTION
4. REACTOR WATER CLEANUP SYSTEM ISOLATION (Continued)				
f. Main Steam Line Tunnel Ambient Temperature - High	8	1	1, 2, 3	27
g. Main Steam Line Tunnel Δ Temp. - High	8	1	1, 2, 3	27
h. SLCS Initiation	8(i)	~~NA~~ 1	1, 2, ~~3~~ 5##	~~27~~ 30
i. Manual Initiation	8	2	1, 2, 3	26
5. REACTOR CORE ISOLATION COOLING SYSTEM ISOLATION				
a. RCIC Steam Line Flow - High	4	1	1, 2, 3	27
b. RCIC Steam Supply Pressure - Low	4, 9(m)	1	1, 2, 3	27
c. RCIC Turbine Exhaust Diaphragm Pressure - High	4	2	1, 2, 3	27
d. RCIC Equipment Room Ambient Temperature - High	4	1	1, 2, 3	27
e. RCIC Equipment Room Δ Temp. - High	4	1	1, 2, 3	27
f. Main Steam Line Tunnel Ambient Temperature - High	4	1	1, 2, 3	27
g. Main Steam Line Tunnel Δ Temp. - High	4	1	1, 2, 3	27
h. Main Steam Line Tunnel Temperature Timer	4	1	1, 2, 3	27

TABLE 3.3.2-1 (Continued)
ISOLATION ACTUATION INSTRUMENTATION

ACTION

ACTION 20 - Be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.

ACTION 21 - Close the affected system isolation valve(s) within one hour or:
- a. In OPERATIONAL CONDITION 1, 2, or 3, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.
- b. In Operational Condition *, suspend CORE ALTERATIONS, handling of irradiated fuel in the primary containment and operations with a potential for draining the reactor vessel.

ACTION 22 - Restore the manual initiation function to OPERABLE status within 48 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

ACTION 23 - Be in at least STARTUP with the associated isolation valves closed within 6 hours or be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.

ACTION 24 - Be in at least STARTUP within 6 hours.

ACTION 25 - Establish SECONDARY CONTAINMENT INTEGRITY with the standby gas treatment system operating within one hour.

ACTION 26 - Restore the manual initiation function to OPERABLE status within 8 hours or close the affected system isolation valves within the next hour and declare the affected system inoperable.

ACTION 27 - Close the affected system isolation valves within one hour and declare the affected system inoperable.

ACTION 28 - Lock the affected system isolation valves closed within one hour and declare the affected system inoperable.

ACTION 29 - Close the affected system isolation valves within one hour and declare the affected system or component inoperable or:
- a. In OPERATIONAL CONDITION 1, 2 or 3 be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.
- b. In OPERATIONAL CONDITION # suspend CORE ALTERATIONS and operations with a potential for draining the reactor vessel.

ACTION 30 - DECLARE THE AFFECTED SLCS PUMP INOPERABLE.

NOTES

* When handling irradiated fuel in the primary or secondary containment and during CORE ALTERATIONS and operations with a potential for draining the reactor vessel.

** The low condenser vacuum MSIV closure may be manually bypassed during reactor SHUTDOWN or for reactor STARTUP when condenser vacuum is below the trip setpoint to allow opening of the MSIVs. The manual bypass shall be removed when condenser vacuum exceeds the trip setpoint.

\# During CORE ALTERATIONS and operations with a potential for draining the reactor vessel.

(a) See Specification 3.6.4, Table 3.6.4-1 for valves in each valve group.

(b) A channel may be placed in an inoperable status for up to 2 hours for required surveillance without placing the trip system in the tripped condition provided at least one other OPERABLE channel in the same trip system is monitoring that parameter.

(c) Also actuates the standby gas treatment system.

(d) Also actuates the control room emergency filtration system in the isolation mode of operation.

(e) Two upscale-Hi Hi, one upscale-Hi Hi and one downscale, or two downscale signals from the same trip system actuate the trip system and initiate isolation of the associated containment and drywell isolation valves.

\#\# WITH ANY CONTROL ROD WITHDRAWN. NOT APPLICABLE TO CONTROL RODS REMOVED PER SPECIFICATION 3.9.10.1 OR 3.9.10.2.

TABLE 4.3.2.1-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION		CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	OPERATIONAL CONDITIONS IN WHICH SURVEILLANCE REQUIRED
4.	REACTOR WATER CLEANUP SYSTEM ISOLATION (Continued)				
f.	Main Steam Line Tunnel Ambient Temperature - High	S	M	R	1, 2, 3
g.	Main Steam Line Tunnel Δ Temp. - High	S	M	R	1, 2, 3
h.	SLCS Initiation	NA	M(b)	NA	1, 2, ~~3~~ 5##
i.	Manual Initiation	NA	M(a)	NA	1, 2, 3
5.	REACTOR CORE ISOLATION COOLING SYSTEM ISOLATION				
a.	RCIC Steam Line Flow - High	S	M	R	1, 2, 3
b.	RCIC Steam Supply Pressure - Low	S	M	R	1, 2, 3
c.	RCIC Turbine Exhaust Diaphragm Pressure - High	S	M	R	1, 2, 3
d.	RCIC Equipment Room Ambient Temperature - High	S	M	R	1, 2, 3
e.	RCIC Equipment Room Δ Temp. - High	S	M	R	1, 2, 3
f.	Main Steam Line Tunnel Ambient Temperature - High	S	M	R	1, 2, 3
g.	Main Steam Line Tunnel Δ Temp. - High	S	M	R	1, 2, 3

TABLE 4.3.2.1-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	OPERATIONAL CONDITIONS IN WHICH SURVEILLANCE REQUIRED
6. RHR SYSTEM ISOLATION (Continued)				
e. Drywell Pressure - High	S	M	R	1, 2, 3
f. Manual Initiation	NA	M(a)	NA	1, 2, 3

*When handling irradiated fuel in the primary or secondary containment and during CORE ALTERATIONS and operations with a potential for draining the reactor vessel.

**The low condenser vacuum MSIV closure may be manually bypassed during reactor SHUTDOWN or for reactor STARTUP when condenser vacuum is below the trip setpoint to allow opening of the MSIVs. The manual bypass shall be removed when condenser vacuum exceeds the trip setpoint.

#During CORE ALTERATION and operations with a potential for draining the reactor vessel.

(a) Manual initiation switches shall be tested at least once per 18 months during shutdown. All other circuitry associated with manual initiation shall receive a CHANNEL FUNCTIONAL TEST at least once per 31 days as part of circuitry required to be tested for automatic system isolation.

(b) Each train or logic channel shall be tested at least every other 31 days.

(c) Calibrate trip unit at least once per 31 days.

WITH ANY CONTROL ROD WITHDRAWN. NOT APPLICABLE TO CONTROL RODS REMOVED PER SPECIFICATION 3.9.10.1 OR 3.9.10.2.