Attachment 1

Proposed Amendment to McGuire Units 1 and 2

Technical Specification 3.5.1.2 Concerning the

Upper Head Injection Accumulator System

EMERGENCY CORE COOLING SYSTEMS

UPPER HEAD INJECTION

LIMITING CONDITION FOR OPERATION

3.5.1.2 Each Upper Head Injection Accumulator System shall be OPERABLE with:

a. The isolation valves open,

b. The water-filled accumulator containing a minimum of 1850 cubic feet of borated water having a concentration of between 1900 and 2100 ppm of boron, and

c. The nitrogen bearing accumulator pressurized to between 1206 and 1264 psig.

APPLICABILITY: MODES 1, 2 and 3.*

ACTION:

Above 46% RATED THERMAL POWER:

a. With the Upper Head Injection Accumulator System inoperable, except as a result of a closed isolation valve(s), restore the Upper Head Injection Accumulator System to OPERABLE status within 1 hour or be at less than or equal to 46% RATED THERMAL POWER and close the isolation valves within the next 6 hours.

b. With the Upper Head Injection Accumulator System inoperable due to the isolation valve(s) being closed, either immediately open the isolation valve(s) or be at less than or equal to 46% RATED THERMAL POWER and close the remaining isolation valves within 1 hour.

Less than or equal to 46% RATED THERMAL POWER:

a. With the Upper Head Injection Accumulator System inoperable, POWER OPERATION may continue provided the isolation valves are closed within 6 hours.

b. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.5.1.2 Each Upper Head Injection Accumulator System shall be demonstrated OPERABLE:

a. At least once per 12 hours by:

1) Verifying the contained borated water volume and nitrogen pressure in the accumulators, and

2) Verifying that each accumulator isolation valve is open.

*Pressurizer Pressure above 1900 psig.

Justification and Safety Analysis

The proposed amendments would revise Technical Specification 3.5.1.2 to allow operation at less than or equal to 46% Rated Thermal Power (RTP) with the Upper Head Injection Accumulator System (UHI) inoperable. A large and small break ECCS analysis has been performed to verify that the appropriate safety parameters would not exceed their limits if an accident occurred during operation under these conditions.

A large and small break ECCS LOCA analysis has been performed for McGuire Units 1 and 2 to determine what power level can be run with the UHI accumulator valved out of service. Tech. Spec. limits on Fq were maintained and the cold leg accumulators parameters were not changed. The analysis shows that the McGuire units can operate at 1569 MW (46% of Rated Thermal Power) with the UHI System out of service. The limiting break was a C_D = 0.8 DECLG which resulted in a Peak Clad Temperature (PCT) of 2102°F which is less than the 2200°F limit established by 10 CFR 50.46. The results for the break spectrum are shown in Table 1. The limiting small break was a six inch cold leg break which resulted in a PCT of 1576°F (1705.5 MW or 50 percent of RTP). The results for the spectrum are shown in Table 2.

The large break analysis was performed with the 1981 ECCS evaluation model. UHI guide tubes and support columns were modelled to provide for realistic draining of the upper head. The UHI accumulator was assumed to be out of service and the cold leg accumulators remained in their UHI mode of 400 psi N_2 and flow restrictors in the delivery lines.

The analysis was performed at reduced power by the Tech. Spec. limits on core power distribution remained in effect. That is, for the 100 percent to 50 percent power range the F_q^t was varied such that the product of F_q^t times power remained constant. (F_q = 2.32/P, P = power/3411 MW). Below 50 percent power the F_q^t remained constant at a maximum value of 4.64.

In the 45 percent to 50 percent power range and a F_q^t = 4.64, a chopped cosine power shape is not possible. A modified power shape was therefore produced which had a peak at 6.0 ft, was symmetric about 6.0 ft and maintained a high percentage of power in the 4.5 ft to 7.5 ft range.

The small break analysis was performed using the 1975 evaluation model. UHI guide tubes and support columns were modelled.

During operation with the UHI system inoperable, the isolation valves must be closed to avoid the possibility of injecting nitrogen into the reactor coolant if the isolation valves are not capable of closing automatically.

Attachment 1

TABLE 1A

LARGE BREAK RESULTS

Results	CD = 0.6 DECL*	CD = 0.8 DECL	CD = 1.0 DECL
Peak Clad Temp. °F	2080	2102	2063
Peak Clad Location Ft.	7.0	7.0	7.0
Local Zr/H_2O Reaction (max)%	2.33	4.98	1.95
Local Zr/H_2O Location Ft.	6.75	7.0	6.75
Total Zr/H_2O Reaction %	<0.3	<0.3	<0.3
Hot Rod Burst Time sec.	102.9	80.1	86.2
Hot Rod Burst Location Ft.	6.25	6.0	6.0

Calculation Assumptions	
Core Power, Mwt, 102% of	1569 MW
Peak Linear Power kw/ft 102% of	11.62
Peaking Factor	4.64
Accumulator Water Volume (Cold Leg Nominal Setpoint Value)	1120 ft^3 per accumulator
Accumulator Water Volume (UHI, Nominal Delivered Value)	Valved Out

* The C_D = 0.6 results are for 48% power (1637 MW)

Attachment 1

TABLE 1B

LB TIME SEQUENCE OF EVENTS

Time in Seconds	CD = 0.6 DECL	CD = 0.8 DECL	CD = 1.0 DECL (Sec.)
START	0.0	0.0	0.0
Reactor Trip Signal	2.3	2.3	2.3
Safety Injection Signal	2.9	2.9	2.9
End of Blowdown	39.8	31.2	28.3
Bottom of Core Recovery	65.4	56.7	51.0
Pump Injection	27.9	27.9	27.9
Accumulator Injection (Cold Leg)	18.8	15.7	14.5
End of Bypass	38.2	31.1	26.3
Accumulator Empty (Cold Leg)	168.9	162.6	160.4

Attachment 1

TABLE 2A

SMALL BREAK RESULTS

Case Analysis Results	4.0 Inch	6.0 Inch	8.0 Inch
Peak Clad Temp. (F)	1199	1576	1232
Peak Clad Location (Ft.)	11.0	11.25	11.0
Local Zr/H_2O Reaction (Max)%	0.374	0.510	0.374
Local Zr/H_2O Location (Ft.)	12.0	12.0	12.0
Total Zr/H_2O Reaction %	<0.3	<0.3	<0.3
Hot Rod Burst Time (Sec)	N/A	N/A	N/A
Hot Rod Burst Location (Ft.)	N/A	N/A	N/A

CALCULATIONAL ASSUMPTIONS

NSSS Power (Hydraulic Analysis), 102% OF	1705.5 MW
Core Power (Rod Heatup Calculation), 102% OF	1705.5 MW
Peak Linear Power kw/ft 102% OF	12.63
Peaking Factor (At 1705.5 MW)	4.64
Cold Leg Accumulator Water Volume (Nominal Setpoint Value)	1120 ft^3
UHI Accumulator Water Volume (Nominal Delivered Value)	Valved Out

Attachment 1

TABLE 2B

SMALL BREAK-TIME SEQUENCE OF EVENTS

Case Analysis (All Times in Seconds)	4.0 Inch	6.0 Inch	8.0 Inch
Start	0.0	0.0	0.0
Reactor Trip Signal Scram Time	14.5	9.2	7.6
Pumped ECCS Injection Begins	39.1	31.2	29.0
Top of Core Uncovered	466	191	112
PCT Occurs	582	519	296
Top of Core Covered	588	445	300
Cold Leg Accumulator Injection Begins	1038.8	397.8	219.4

Analysis of Significant Hazards Consideration

This analysis is provided as required by 10 CFR 50.91 and is performed according to the standards given in 10 CFR 50.92.

The probability of an accident previously evaluated is unaffected by the proposed amendments because the UHI System serves only to mitigate accidents after they occur and performs no function during normal operation. The consequences of an accident previously evaluated are not increased because the most limiting ECCS analysis case previously evaluated (Double-ended cold-leg guillotine break, C_D = 0.6, perfect mixing) remains the most limiting. Furthermore, the results of the analysis provided with the amendment application show that the limiting parameter, peak cladding temperature, would not exceed the limit established by 10 CFR 50.46.

The proposed amendments would not create the possibility of a new or different type of accident from any accident previously evaluated. In addition, the proposed amendments do not involve a significant change in safety margins because the safety margins inherent in the peak cladding temperature limit of 2200°F and in the conservative assumptions of the accident analyses are unaffected.

The Commission has provided guidance concerning the application of standards of no significant hazard determination by providing certain examples (48 FR 14870). One of the examples of actions likely to involve no significant hazards considerations relates to 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. Because the analysis described in the application for the proposed amendments shows that the results of the changes are clearly within the applicable acceptance criteria, the example described above can be applied to this situation.

For the reasons described above, Duke Power Company concludes that the proposed amendments do not involve significant hazards considerations.

Attachment 2

McGuire Nuclear Station
Units 1 and 2

Proposed Amendment to Technical Specifications to
Allow Changing Between Operational Modes 5 and 6
With the Control Area Ventilation System Inoperable

PLANT SYSTEMS

3/4.7.6 CONTROL AREA VENTILATION SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.6 Two independent Control Area Ventilation Systems shall be OPERABLE.

APPLICABILITY: ALL MODES

ACTION: (Units 1 and 2)

MODES 1, 2, 3 and 4:

With one Control Area Ventilation System inoperable, restore the inoperable system to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

MODES 5 and 6:

a. With one Control Area Ventilation System inoperable, restore the inoperable system to OPERABLE status within 7 days or initiate and maintain operation of the remaining OPERABLE Control Area Ventilation System in the recirculation mode; and

b. With both Control Area Ventilation Systems inoperable, or with the OPERABLE Control Area Ventilation System, required to be in the recirculation mode by ACTION a., not capable of being powered by an OPERABLE emergency power source, suspend all operations involving CORE ALTERATIONS or positive reactivity changes.

c. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.6 Each Control Area Ventilation System shall be demonstrated OPERABLE:

a. At least once per 12 hours, by verifying that the control room air temperature is less than or equal to 120°F;

b. At least once per 31 days on a STAGGERED TEST BASIS, by initiating, from the control room, flow through the HEPA filters and charcoal adsorbers and verifying that the system operates for at least 10 hours with the heaters operating;

Justification and Safety Analysis

The proposed amendments would make specification 3.0.4 not applicable in modes 5 (cold shutdown) and 6 (refueling) for the Control Area Ventilation Systems. This would allow changing between modes 5 and 6 with the systems inoperable.

The Control Area Ventilation Systems ensure that the control room remains habitable after postulated accidents. Changing between modes 5 and 6 with the system(s) inoperable is acceptable for the following reasons:

(1) Whether in mode 5 or 6, there is no significant difference in the probability of a reactor accident occurring for which the system would be required to function. In either case, the reactor is substantially subcooled and subcritical.

(2) The fact that mode 5 is acceptably safe is clear because the ACTION section requires proceeding to mode 5 from higher modes but does not require proceeding to mode 6. Per the ACTION section, the remaining operable system would be placed in the recirculation mode. The ACTION section also restricts positive reactivity changes with both systems inoperable and with emergency power unavailable; however, changing from mode 6 to mode 5 does not necessarily involve positive reactivity changes. Therefore, passage from mode 6 to mode 5 is acceptable.

(3) Because the reactivity and temperature limits for mode 6 are lower than for mode 5, passage into mode 6 does not place the unit in a more degraded condition. Therefore, passage from mode 5 to mode 6 is acceptable.

Analysis of Significant Hazards Consideration

This analysis is provided in accordance with 10 CFR 50.91 and is performed according to the standards of 10 CFR 50.92.

The proposed amendments would not involve a significant increase in the probability of an accident previously evaluated because the Control Area Ventilation System is designed to mitigate the consequences of accidents and can have no effect on cause mechanisms. The consequences of accidents previously evaluated would not be significantly increased because accidents which might occur in modes 5 or 6 would be much less severe than the design basis accidents. Further the ACTION requirements provide for appropriate measures to compensate for the system inoperability (such as placing the remaining operable system in recirculation and suspending core alterations and positive reactivity changes).

The proposed amendments would not create the possibility of a new or different kind of accident than previously evaluated. The Control Area Ventilation System cannot cause an accident to occur. Safety margins are not significantly

reduced by the proposed amendments because the design basis accidents involve initial conditions more severe than those conditions (modes 5 and 6) for which the proposed amendments would apply.

Based on the above analysis, it is concluded that the proposed amendments do not involve significant hazards considerations.

Attachment 3

Proposed Amendment to McGuire Units 1 and 2
Technical Specification 4.6.5.3.1
Concerning Ice Condenser Inlet Door Surveillance

CONTAINMENT SYSTEMS

ICE CONDENSER DOORS

LIMITING CONDITION FOR OPERATION

3.6.5.3 The ice condenser inlet doors, intermediate deck doors, and top deck doors shall be closed and OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

With one or more ice condenser doors open or otherwise inoperable, POWER OPERATION may continue for up to 14 days provided the ice bed temperature is monitored at least once per 4 hours and the maximum ice bed temperature is maintained less than or equal to 27°F; otherwise, restore the doors to their closed positions or OPERABLE status (as applicable) within 48 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.5.3.1 Inlet Doors - Ice condenser inlet doors shall be:

a. Continuously monitored and determined closed by the inlet door position monitoring system, and

b. Demonstrated OPERABLE during shutdown at least once per ~~3~~ 9 months ~~during the first year after the ice bed is fully loaded and at least once per 6 months thereafter~~ by:

   1) Verifying that the torque required to initially open each door is less than or equal to 675 inch pounds;

   2) Verifying that opening of each door is not impaired by ice, frost or debris;

   3) Testing a sample of at least ~~25%~~ 50% of the doors and verifying that the torque required to open each door is less than 195 inch-pounds when the door is 40 degrees open. This torque is defined as the "door opening torque" and is equal to the nominal door torque plus a frictional torque component. The doors selected for determination of the "door opening torque" shall be selected to ensure that all doors are tested at least once during ~~four~~ two test intervals;

CONTAINMENT SYSTEMS

SURVEILLANCE REQUIREMENTS (Continued)

4) Testing a sample of at least ~~25%~~ 50% of the doors and verifying that the torque required to keep each door from closing is greater than 78 inch-pounds when the door is 40 degrees open. This torque is defined as the "door closing torque" and is equal to the nominal door torque minus a frictional torque component. The doors selected for determination of the "door closing torque" shall be selected to ensure that all doors are tested at least once during ~~four~~ two test intervals; and

5) Calculation of the frictional torque of each door tested in accordance with 3) and 4), above. The calculated frictional torque shall be less than or equal to 40 inch-pounds.

4.6.5.3.2 Intermediate Deck Doors - Each ice condenser intermediate deck door shall be:

a. Verified closed and free of frost accumulation by a visual inspection at least once per 7 days, and

b. Demonstrated OPERABLE at least once per 3 months during the first year after the ice bed is fully loaded and at least once per 18 months thereafter by visually verifying no structural deterioration, by verifying free movement of the vent assemblies, and by ascertaining free movement when lifted with the applicable force shown below:

	Door	Lifting Force
1)	Adjacent to crane wall	Equal to or less than 37.4 lbs,
2)	Paired with door adjacent to crane wall	Equal to or less than 33.8 lbs,
3)	Adjacent to containment wall	Equal to or less than 31.8 lbs, and
4)	Paired with door adjacent to containment wall	Equal to or less than 31.0 lbs.

4.6.5.3.3 Top Deck Doors - Each ice condenser top deck door shall be determined closed and OPERABLE at least once per 92 days by visually verifying:

a. That the doors are in place, and

b. That no condensation, frost, or ice has formed on the doors or blankets which would restrict their lifting and opening if required.

Justification and Safety Analysis

The proposed amendments would increase the surveillance interval for verifying that the ice condenser inlet doors can be opened and closed properly with the specified torque. The proposed amendments would also increase the size of the sample required to be tested during each surveillance.

The surveillance interval would be changed from 6 months (3 months during the first year) to 9 months. Since this testing cannot be performed during unit operation, the existing specification will require a unit outage every 6 months solely to perform this surveillance. Changing the interval to 9 months would allow this testing to coincide with the outage to weigh ice baskets per Specification 4.6.5.1.b.

It is also proposed that the sample size for verifying the "door opening torque" and "door closing torque" be increased from 25% to 50%. By testing a larger sample of doors, the change would result in each door being tested more frequently--at least once per 18 months instead of 24 months under the existing specification--despite the increased surveillance interval.

Justification for the increased surveillance interval is provided by the surveillance history at McGuire. The inlet door surveillance has been performed 10 times over a 2 year period on Unit 1 with no failures. It has also been performed one time on Unit 2 with no failures. This provides substantial confidence that the inlet doors would not develop problems during the proposed 9 month surveillance interval.

One reason for the excellent surveillance history is a design change made to the door seals to prevent the doors from freezing closed. The old seal design allowed condensation to collect at the seals and freeze the doors closed. This problem was experienced at another plant. The solution which was implemented at McGuire before initial startup was to redesign the door seals to prevent condensation from collecting.

Analysis Related to Significant Hazards Consideration

10 CFR 50.92 states that a proposed amendment involves no significant hazards considerations if operation in accordance with the proposed amendment would not:

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

(2) Create the possibility of a new or different kind of accident from any accident previously evaluated; or

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

Because the function of the ice condenser inlet doors is to mitigate the consequences of accidents, the proposed changes can have no effect on the probability of those accidents. The changes will result in each door having its "door opening torque" and "door closing torque" (as defined in Specification 4.6.5.3.1.b)

verified more frequently--every 18 months instead of 24 months. Also, the torque required to initially open each door would be verified less frequently--every 9 months instead of 6 months. However, the acceptable criteria for the tests will not be changed. Therefore, the net effect of the changes, in terms of level of confidence that the doors will perform acceptably, is not significantly affected. Because the ice condenser inlet doors can be expected to perform acceptably based upon the proposed surveillance requirements, the consequences of accidents are not increased.

The proposed amendments involve changes in surveillance frequency and do not involve changes in surveillance methods, acceptance criteria, nor operating conditions. Therefore, no new or different kinds of accidents are created.

The margins of safety associated with the ice condenser inlet doors are contained within the containment pressure accident analysis and the resulting surveillance acceptance criteria. Since no change to the accident analysis nor the surveillance acceptance criteria is proposed, the proposed changes will not affect the margins of safety.

Based upon the above analysis, the proposed amendments do not involve significant hazards considerations.

Attachment 4

Proposed Amendment to McGuire Units 1 and 2
Technical Specifications Table 3.3-5
Concerning Steam Line Isolation Response Time

TABLE 3.3-5 (Continued)

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
3. Pressurizer Pressure-Low-Low	
a. Safety Injection (ECCS)	≤ 27(1)/12(3)
b. Reactor Trip (from SI)	≤ 2
c. Feedwater Isolation	≤ 9
d. Containment Isolation-Phase "A"(2)	≤ 18(3)/28(4)
e. Containment Purge and Exhaust Isolation	N.A.
f. Auxiliary Feedwater(5)	N.A.
g. Nuclear Service Water System	≤ 76(1)/65(3)
h. Component Cooling Water	≤ 76(1)/65(3)
i. Start Diesel Generators	≤ 11
4. Steam Line Pressure-Low	
a. Safety Injection (ECCS)	≤ 12(3)/22(4)
b. Reactor Trip (from SI)	≤ 2
c. Feedwater Isolation	≤ 9
d. Containment Isolation-Phase "A"(2)	≤ 18(3)/28(4)
e. Containment Purge and Exhaust Isolation	N.A.
f. Auxiliary Feedwater(5)	N.A.
g. Nuclear Service Water	≤ 65(3)/76(4)
h. Steam Line Isolation	≤ ~~9~~ 7
i. Component Cooling Water	≤ 65(3)/76(4)
j. Start Diesel Generators	≤ 11
5. Containment Pressure--High-High	
a. Containment Spray	≤ 45
b. Containment Isolation-Phase "B"	N.A.
c. Steam Line Isolation	≤ 9 7
6. Steam Generator Water Level--High-High	
a. Turbine Trip	N.A.
b. Feedwater Isolation	≤ 13

TABLE 3.3-5 (Continued)

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
7. Steam Generator Water Level - Low-Low	
a. Motor-driven Auxiliary Feedwater Pumps	≤ 60
b. Turbine-driven Auxiliary Feedwater Pumps	≤ 60
8. Negative Steam Line Pressure Rate - High	
Steam Line Isolation	≤ ~~9~~ 7
9. Start Permissive	
Containment Pressure Control System	N.A.
10. Termination	
Containment Pressure Control System	N.A.
11. Auxiliary Feedwater Suction Pressure - Low	
Auxiliary Feedwater Pumps (Suction Supply Automatic Realignment)	≤ 13
12. RWST Level	
Automatic Switchover to Recirculation	≤ 60
13. Station Blackout	
a. Start Motor-Driven Auxiliary Feedwater Pumps	≤ 60
b. Start Turbine-Driven Auxiliary Feedwater Pump	≤ 60
14. Trip of Main Feedwater Pumps	
Start Motor-Driven Auxiliary Feedwater Pumps	≤ 60
15. Loss of Power	
4 kV Emergency Bus Undervoltage-Grid Degraded Voltage	≤ 11

Justification and Safety Analysis

The proposed amendments would change the required response time for steam line isolation from $\leq$ 9 seconds to $\leq$ 7 seconds. This change is necessary because both the FSAR Chapter 6 steamline break containment analysis and the Chapter 15 steamline break core analysis assumed a steamline isolation response time of 7 seconds. The reason for the discrepancy between these analyses and the current Technical Specification limit is not known; however, a review of all previous response time tests showed that the 7 seconds response time was met in each case. Pending issuance of the proposed amendments, the 7 seconds response time limit is being observed. The proposed amendments would involve a more restrictive limit than the current limit and is therefore acceptable.

Analysis of Significant Hazards Considerations

Because the proposed amendments involve operation under a more restrictive requirement than was previously approved, the proposed amendments do not involve significant hazards considerations according to the standards of 10 CFR 50.92

Attachment 5

Proposed Amendments to McGuire Units 1 and 2
Technical Specifications Table 3.7-4b Concerning Snubbers

PLANT SYSTEMS

TABLE 3.7-4b (Continued)

SAFETY-RELATED MECHANICAL SNUBBERS*

PACIFIC SCIENTIFIC

SYSTEM**	SMALL SIZE (350 Lbs. or Less TO 600 Lbs)	MEDIUM SIZE (1,487 Lbs. to 15,000 Lbs)	LARGE SIZE (50,000 Lbs. to 120,000 Lbs)
SV	0	3	0
VE	1	3	0
VI	30	1	0
VQ	0	2	0
WG	2	0	0
WL	8	0	0
WS	2	0	0
YC	1	1	0
Subtotal (Unit 1)	326	282	23
UNIT 2			
BB	67	55	0
CA	9	59	0
CF	13	80	8
FW	0	4	1
KC	66	81	0
KD	3	1	0
KF	2	5	0
LD	0 ~~1~~	2	0
NB	5	1	0
NC	100	95	2
ND	29	41	0

*Snubbers may be added or deleted without prior License Amendment to Table 3.7-4b provided that a revision to Table 3.7-4b is included with the next License Amendment request. In lieu of any other report required by Specification 6.9.1, at least 15 days prior to the deletion of any listed snubber, a Special Report shall be prepared and submitted to the Commission in accordance with Specification 6.9.2 evaluating the safety significance of the proposed snubber removal.

**A listing of individual snubbers and more detailed information shall be available for NRC review at the McGuire Nuclear Station.

TABLE 3.7-4b (Continued)

SAFETY-RELATED MECHANICAL SNUBBERS*

PACIFIC SCIENTIFIC

SYSTEM**	SMALL SIZE (350 Lbs. or Less TO 600 Lbs)	MEDIUM SIZE (1,487 Lbs. to 15,000 Lbs)	LARGE SIZE (50,000 Lbs. to 120,000 Lbs)
NF	2	2	0
NI	56	~~56~~ 55	3
NM	42	11	0
NR	10	8	0
NV	170	69	0
RF	1	3	0
RN	28	34	1
RV	9	8	0
SA	9	12	0
SM	2	24	30
SV	0	1	0
TE	1	3	0
VE	3	2	0
VG	1	2	0
VI	26	1	0
VN	0	4	0
VQ	3	1	0
VS	1	0	0
VX	2	1	0
WL	15	7	0
WN	0	2	0
Subtotal (Unit 2)	~~676~~ 675	~~675~~ 674	45
TOTAL for UNITS 1 and 2	~~1,002~~ 1,001	~~957~~ 956	65

*Snubbers may be added or deleted without prior License Amendment to Table 3.7-4b provided that a revision to Table 3.7-4b is included with the next License Amendment request. In lieu of any other report required by Specification 6.9.1, at least 15 days prior to the deletion of any listed snubber, a Special Report shall be prepared and submitted to the Commission in accordance with Specification 6.9.2 evaluating the safety significance of the proposed snubber removal.

**A listing of individual snubbers and more detailed information shall be available for NRC review at the McGuire Nuclear Station.

Justification and Safety Analysis

The proposed amendments would involve revising Technical Specifications Table 3.7-4b to reflect the deletion of one small-size mechanical snubber on the Unit 2 Diesel Generator Lube Oil (LD) System and one medium size mechanical snubber on the Unit 2 Safety Injection (NI) System. Snubber No. 2-MCA-LD-3019 is located in the Diesel Generator 2A pipe trench and is difficult to access for maintenance and inspections. The piping math model was reanalyzed without this snubber according to the ASME code requirements for Duke Class C piping. The results of the analysis showed that allowable stresses can be met without the snubber.

Snubber No. 2 MCR-NI-4945 which is part of the Safety Injection System will be replaced with a rigid support. The rigid support will be designed to withstand seismic loading as necessary to compensate for the deleted snubber. Stresses caused by thermal movements have been analyzed and determined to be within acceptable limits. Therefore, this modification will have no safety significance.

The modifications to delete the snubbers might be implemented prior to issuance of the proposed amendments as allowed by the Technical Specification. Please note that Special Reports were submitted in accordance with the Technical Specification at least 15 days prior to the modifications.

Analysis of Significant Hazards Considerations

This analysis is provided as required by 10 CFR 50.91 and is performed according to the standards of 10 CFR 50.92.

Because the reanalysis of the piping math model has shown that ASME code requirements can be met without the small snubber on the Unit 2 LD System and because a rigid support will be added to compensate for the deletion of the snubber on the Unit 2 NI system, the proposed amendments would not:

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

(2) Create the possibility of a new or different kind of accident from any accident previously evaluated; or

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

Therefore, the proposed amendments do not involve significant hazards considerations.

Attachment 6

Proposed Amendments to McGuire Units 1 and 2
Technical Specification 4.6.1.1.a Concerning
Verification of Primary Containment Integrity

3/4.6 CONTAINMENT SYSTEMS

3/4.6.1 PRIMARY CONTAINMENT

CONTAINMENT INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.1.1 Primary CONTAINMENT INTEGRITY shall be maintained.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

Without primary CONTAINMENT INTEGRITY, restore CONTAINMENT INTEGRITY within 1 hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.1 Primary CONTAINMENT INTEGRITY shall be demonstrated:

a. At least once per 31 days by verifying that all penetrations* not capable of being closed by OPERABLE containment automatic isolation valves and required to be closed during accident conditions are closed by valves, blind flanges, or deactivated automatic valves secured in their positions, except as provided in Table 3.6-2 of Specification 3.6.3;

b. By verifying that each containment air lock is in compliance with Specification 3.6.1.3; and

c. After each closing of each penetration subject to Type B testing, except the containment air locks, if opened following a Type A or B test, by leak rate testing the seal with gas at P_a, 14.8 psig, and verifying that when the measured leakage rate for these seals is added to the leakage rates determined pursuant to Specification 4.6.1.2d. for all other Type B and C penetrations, the combined leakage rate is less than 0.60 L_a.

and the annulus

*Except valves, blind flanges, and deactivated automatic valves which are located inside the containment and are locked, sealed or otherwise secured in the closed position. These penetrations shall be verified closed during each COLD SHUTDOWN except that such verification need not be performed more often than once per 92 days.

Justification and Safety Analysis

The proposed amendments would revise the footnote to Technical Specification 4.6.1.1.a to exempt locked valves, blind flanges, and deactivated automatic valves located inside the annulus from the monthly surveillance requirements of 4.6.1.1.a. This exemption already applies to components inside containment. Surveillance would be performed during cold shutdown as required for components inside containment.

The purpose of the proposed amendments is to avoid access to the annulus during operation to reduce radiation exposure to personnel. Portions of the annulus are considered high radiation levels during operation. This exemption was previously approved by the NRC staff for the Unit 1 Technical Specifications; however, it was inadvertently omitted when developing the Units 1 and 2 Technical Specifications.

Analysis of Significant Hazards Considerations

This analysis is provided as required by 10 CFR 50.91 and is performed according to the standards of 10 CFR 50.92.

The proposed amendments would involve less frequent surveillance of the status of penetrations in the annulus. Because these penetrations are locked, sealed, or otherwise secured in the closed position, they can only be repositioned by personnel error. This is unlikely because access to the annulus during operation is restricted except for essential tasks. Therefore, the effect of the proposed change is relatively insignificant to safety.

The Commission has provided guidance concerning the application of standards of no significant hazard determination by providing certain examples (48 FR 14870). One of the examples of actions likely to involve no significant hazards considerations relates to 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. Section 6.2.4 of the Standard Review Plan (SRP), paragraph II.3.f provides criteria for seal closed barriers defined as including locked closed valves, blind flanges, and closed automatic valves which remain closed after a LOCA. While the SRP provides criteria for administrative controls, it does not address the frequency of verification that administrative controls remain in place. Since the results of the proposed amendments are clearly within all acceptable criteria for the components specified in the SRP and was previously approved by the NRC staff for McGuire Unit 1, the example described above applies.

Therefore, the proposed amendments do not involve significant hazards considerations.

Attachment 7

Proposed Amendment to McGuire Units 1 and 2
Technical Specification 4.6.1.3.b Concerning
Containment Airlock Surveillance

CONTAINMENT SYSTEMS

SURVEILLANCE REQUIREMENTS

4.6.1.3 Each containment air lock shall be demonstrated OPERABLE:

a. Within 72 hours following each closing, except when the air lock is being used for multiple entries, then at least once per 72 hours, by verifying that the seal leakage is less than 0.01 L_a as determined by precision flow measurements when measured for at least 30 seconds with the volume between the seals at a constant pressure of 14.8 psig,

b. By conducting overall air lock leakage tests at not less than P_a, 14.8 psig, and verifying the overall air lock leakage rate is within its limit:

   1) At least once per 6 months, # and

   2) Prior to establishing CONTAINMENT INTEGRITY when maintenance has been performed on the air lock that could affect the air lock sealing capability.*

c. At least once per 6 months by verifying that only one door in each air lock can be opened at a time, and

d. At least once per 6 months by conducting a pressure test to verify door seal integrity, with a measured leak rate of less than 15 standard cubic centimeters per minute.

The provisions of Specification 4.0.2 are not applicable.

* This constitutes an exemption to Appendix J of 10 CFR 50.

Justification and Safety Analysis

McGuire Units 1 and 2 Technical Specification 4.6.1.3.b currently requires overall containment airlock leakage tests to be performed "...if opened when CONTAINMENT INTEGRITY was not required..." The proposed amendments would change this to require the overall airlock leakage test to be performed "...when maintenance has been performed on the air lock that could affect the air lock sealing capability." This proposed change constitutes an exemption to Appendix J of 10 CFR 50.

The proposed amendments are justified for several reasons:

(1) Opening the airlock, i.e. opening both doors simultaneously, is no different in terms of capability to reseal than opening one door at a time during normal entries.

(2) Test data taken on Unit 1 on sixteen different occasions since June, 1981 has not indicated any tendency of the airlock leakage rate to increase after opening both airlock doors simultaneously. (See the attached table.)

(3) The overall airlock leakage rate will be measured at least once per 6 months regardless of the airlock operating or maintenance history. Also, the test would be performed after maintenance activities potentially affecting the airlock sealing capability.

(4) A seal integrity test is performed prior to establishing containment integrity and once every 72 hours per Specification 4.6.1.3.a. This is a more meaningful and more conservative test for detecting seal problems than the overall airlock leakage rate test because it verifies the integrity of each seal on each door. Because the overall airlock leakage test involves pressurizing between the doors, this test only verifies that at least one of the two seals on each door is sealed. (The McGuire airlocks have four seals between containment and outside.) The overall airlock leakage rate test might detect potential problems with the airlock not related to the door seals; however, such problems would not occur as a result of opening both doors simultaneously.

(5) Meeting the current requirement is a significant burden. The airlocks are usually opened during outages to facilitate equipment transport into and out of containment. Then just prior to entry into mode 4, the overall airlock leakage test must be performed. Installing strongbacks, performing the test, and removing strongbacks requires at least 6 hours per airlock during which access through the airlock is prohibited. Any access and egress to lower containment during testing of the lower airlock involves climbing through the emergency hatch between upper and lower containment. This results in more contamination in upper containment which is usually cleaner than lower containment, Similarly, access

to upper containment while testing the upper airlock requires passing through lower containment where radiation levels are higher, thus increasing radiation exposure to personnel and increasing contamination in upper containment. The proposed changes would allow better scheduling of the overall airlock leakage test during periods when the need for access to containment is minimal.

Analysis of Significant Hazards Considerations

The proposed amendments would remove the requirement to test the overall airlock leakage after each opening of the airlock and add a requirement to test whenever the airlock sealing capability might have been affected by maintenance. Because any effect on the airlock sealing capability potentially caused by opening the doors would be detected by another required test and because the overall airlock leakage test will continue to be performed every 6 months and after maintenance, the proposed changes are insignificant to safety. Therefore, the proposed amendments would not:

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

(2) Create the possibility of a new or different kind of accident from any accident previously evaluated; or

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

Therefore, according to the standards of 10 CFR 50.92, the proposed changes do not involve significant hazards considerations.

McGuire Unit 1
Containment Air Lock Leakage Data

The following test data from June, 1981 to April, 1983 shows the containment air lock leakage rates measured after the air locks had been opened. Note that all tests met the acceptance criterion of 0.05 L_a (4530 sccm.)

Upper Airlock		Lower Airlock	
Date	Leakage (sccm)	Date	Leakage (sccm)
4/23/83	580	4/25/83	655
11/18/82	412	11/20/82	410
7/13/82	290	7/14/82	115
3/12/82	865	3/13/82	225
12/29/81	193	12/28/81	417
11/20/81	258	11/22/81	751
6/06/81	45	10/02/81	492
		8/03/81	951
		6/11/81	259

Attachment 8

Proposed Amendments to McGuire Units 1 and 2
Technical Specifications Concerning
Secondary Containment Bypass Leakage Paths

TABLE 3.6-1 (Continued)

SECONDARY CONTAINMENT BYPASS LEAKAGE PATHS

PENETRATION NUMBER	SERVICE	RELEASE LOCATION	TEST TYPE
M317	Instrument Air	Auxiliary Building	Type C
M243	Containment Air Release	Auxiliary Building	Type C
M384	Containment Air Addition	Auxiliary Building	Type C
M361	Reactor Coolant Pump Motor Oil Supply	Auxiliary Building	Type C
M353	Fire Protection Header	Auxiliary Building	Type C
M376	Component Cooling Water to Reactor Coolant Drain Tank Heat Exchanger	Auxiliary Building	Type C
M355	Component Cooling Water from Reactor Coolant Drain Tank Heat Exchanger	Auxiliary Building	Type C
M327	Component Cooling Water to Reactor Vessel Support Coolers and RCP Coolers	Auxiliary Building	Type C
M320	Component Cooling Water from Reactor Vessel Support Coolers and RCP Coolers	Auxiliary Building	Type C
---	Flued Head to Guard Pipe Welds on all Hot Penetrations	Atmosphere, or Auxiliary Building, or Turbine Building	*
M412	Equipment Hatch	Atmosphere	Type C

*Pursuant to Specification 4.6.1.2f.

TABLE 3.6-1 (Continued)

SECONDARY CONTAINMENT BYPASS LEAKAGE PATHS

PENETRATION NUMBER	SERVICE	RELEASE LOCATION	TEST TYPE
M280	Sample from Accumulator	Auxiliary Building	Type C
M342	Auxiliary Seal Injection Line from Annulus to Reactor Coolant Pumps	Auxiliary Building	Type C
M394	Ice from Rotary Valve Assembly to Ice Condenser Cyclone Receiver	Auxiliary Building	Type C
M255	ILRT Pressure Impulse Line	Auxiliary Building	Type C
M323	Cont. Rad. Monitors EMF-38, 39, 40	Auxiliary Building	Type C
M118	Cont. Press Monitors	Auxiliary Building	Type C
M118	ILRT Press Impulse Line (Unit 2)	Auxiliary Building	Type C
M239	Cont. Press Monitors	Auxiliary Building	Type C
M239	Cont. Hydrogen Monitor "A" Train	Auxiliary Building	Type C
M313	Cont. Press Monitors	Auxiliary Building	Type C
M402	Cont. Press Monitors	Auxiliary Building	Type C
---	Cont. Hydrogen Monitor "B" Train	Auxiliary Building	Type C
M392	Air to Upper PAL Aux Bldg Side Door Seals	Auxiliary Building	Type C
M 152	Air to Lower PAL Aux Bldg Side Door Seals	Auxiliary Building	Type C

TABLE 3.6-1 (Continued)

SECONDARY CONTAINMENT BYPASS LEAKAGE PATHS

PENETRATION NUMBER	SERVICE	RELEASE LOCATION	TEST TYPE
---	Cont. Press. Monitor - Narrow Range	Auxiliary Building	Type C
M354	Fuel Transfer Tube	Auxiliary Building	Type B

Justification and Safety Analysis

The proposed amendments would include several additional penetrations in Table 3.6-1, Secondary Containment Bypass Leakage Paths. These penetrations were inadvertently omitted from the table due to administrative errors. All of these penetrations are currently included in the periodic surveillance program.

Analysis of Significant Hazards Consideration

Because the proposed amendments would involve requirements which are clearly more restrictive than the existing requirements, the proposed amendments do not involve significant hazards considerations according to the standards of 10 CFR 50.92.

Attachment 9

Proposed Amendments to McGuire Units 1 and 2
Technical Specifications Concerning
Administrative and Typographical Errors

REACTOR COOLANT SYSTEM

3/4.4.9 PRESSURE/TEMPERATURE LIMITS

REACTOR COOLANT SYSTEM

LIMITING CONDITION FOR OPERATION

3.4.9.1 The Reactor Coolant System (except the pressurizer) temperature and pressure shall be limited in accordance with the limit lines shown on Figures 3.4-2 and 3.4-3 during heatup, cooldown, criticality, and inservice leak and hydrostatic testing with:

a. A maximum heatup of 100°F for Unit 1 and 60°F for Unit 2 in any 1-hour period,

b. A maximum cooldown of 100°F in any 1-hour period, and

c. A maximum temperature change of less than or equal to 10°F in any 1-hour period during inservice hydrostatic and leak testing operations above the heatup and cooldown limit curves.

APPLICABILITY: At all times.

ACTION:

With any of the above limits exceeded, restore the temperature and/or pressure to within the limit within 30 minutes; perform an engineering evaluation to determine the effects of the out-of-limit condition on the structural integrity of the Reactor Coolant System; determine that the Reactor Coolant System remains acceptable for continued operation or be in at least HOT STANDBY within the next 6 hours and reduce the RCS T_{avg} and pressure to less than 200°F and 500 psig, respectively, within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.9.1.1 The Reactor Coolant System temperature and pressure shall be determined to be within the limits at least once per 30 minutes during system heatup, cooldown, and inservice leak and hydrostatic testing operations.

4.4.9.1.2 The reactor vessel material irradiation surveillance specimens shall be removed and examined, to determine changes in material properties, as required by 10 CFR 50, Appendix H in accordance with the schedule in Table 4.4-5. The results of these examinations shall be used to update Figures 3.4-2 and 3.4-3.

TABLE 4.3-9 (Continued)

TABLE NOTATION

*At all times except when the isolation valve is closed and locked.

**During WASTE GAS HOLDUP SYSTEM operation.

(1) The ANALOG CHANNEL OPERATIONAL TEST shall also demonstrate that automatic isolation of this pathway and control room alarm annunciation occurs if any of the following conditions exists:

a. Instrument indicates measured levels above the Alarm/Trip Setpoint,

b. Circuit failure (alarm only), and

c. Instrument indicates a downscale failure (alarm only).

(2) The ANALOG CHANNEL OPERATIONAL TEST shall also demonstrate that control room alarm annunciation occurs if any of the following conditions exists:

a. Instrument indicates measured levels above the Alarm Setpoint,

b. Circuit failure, and

c. Instrument indicates a downscale failure~~, and~~.

(3) The initial CHANNEL CALIBRATION shall be performed using one or more of the reference standards certified by the National Bureau of Standards (NBS) or using standards that have been obtained from suppliers that participate in measurement assurance activities with NBS. These standards shall permit calibrating the system over its intended range of energy and measurement range. For subsequent CHANNEL CALIBRATION, sources that have been related to the initial calibration shall be used.

(4) The CHANNEL CALIBRATION shall include the use of standard gas samples corresponding to alarm setpoints in accordance with the manufacturer's recommendations.

(5) The CHANNEL CALIBRATION shall include the use of standard gas samples in accordance with the manufacturer's recommendations. In addition, a standard gas sample of nominal 4 volume percent oxygen, balance nitrogen, shall be used in the calibration to check linearity of the oxygen analyzer.

PLANT SYSTEMS

Table 3.7-5

FIRE HOSE STATIONS

NO.	LOCATION	ELEVATION (FT)
157	55-FF	695
180	52-CC	716
181	54-GG	716
176	52-CC	716
175	52-CC	716
893	40-AA	733
891	40-CC	733
892	43 ~~45~~/44-DD	733
895	46 BB	733
889	51/52-DD	733
888	52-EE	733
171	54-HH	733
167	51-JJ	733
168	52-MM	733
169	55-NN	733
962	46-CC	750
964	51-CC	750
965	54-BB	750
966	56-DD	750
971	58-BB	750
303	52-GG	750
162	54-LL	750
161	50-MM	750
164	56-QQ	750
974	51-BB	767
191	56-GG	767
184	54-KK	767
186	51-MM	767
158	57-FF	695
183	59-CC/DD	716
182	58-GG	716
178	58/59-MM	716
179	61-LL	733
900	68-AA/BB	733
901	72-DD	733
902	68/69-DD	733
903	72-DD	733
914	66-BB	733
898	61-BB	733
899	66-BB	733
897	60-DD	733
~~906~~ 896	60-EE	733
904	58-CC/DD	733
172	58-HH	733
174	61-JJ/KK	733
170	57-LL	733

PLANT SYSTEMS

3/4.7.12 AREA TEMPERATURE MONITORING

LIMITING CONDITION FOR OPERATION

3.7.12 The temperature of each area shown in Table 3.7-6 shall be maintained within the limits indicated in Table 3.7-6.

APPLICABILITY: Whenever the equipment in an affected area is required to be OPERABLE.

ACTION:

With one or more areas exceeding the temperature limit(s) shown in Table 3.7-7 6

a. For more than 8 hours, prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 30 days providing a record of the amount by which and the cumulative time the temperature in the affected area exceeded its limit and an analysis to demonstrate the continued OPERABILITY of the affected equipment.

b. By more than 30°F, in addition to the Special Report required above, within 4 hours either restore the area to within its temperature limit or declare the equipment in the affected area inoperable.

SURVEILLANCE REQUIREMENTS

4.7.12 The temperature in each of the areas shown in Table 3.7-6 shall be determined to be within its limit at least once per 12 hours.

3/4.10.2 GROUP HEIGHT, INSERTION, AND POWER DISTRIBUTION LIMITS

LIMITING CONDITION FOR OPERATION

3.10.2 The group height, insertion, and power distribution limits of Specifications 3.1.3.1, 3.1.3.5, 3.1.3.6, ~~3.1.3.7~~, 3.2.1, and 3.2.4 may be suspended during the performance of PHYSICS TESTS provided:

a. The THERMAL POWER is maintained less than or equal to 85% of RATED THERMAL POWER, and

b. The limits of Specifications 3.2.2 and 3.2.3 are maintained and determined at the frequencies specified in Specification 4.10.2.2 below.

APPLICABILITY: MODE 1.

ACTION:

With any of the limits of Specification 3.2.2 or 3.2.3 being exceeded while the requirements of Specifications 3.1.3.1, 3.1.3.5, 3.1.3.6, 3.2.1, and 3.2.4 are suspended, either:

a. Reduce THERMAL POWER sufficient to satisfy the ACTION requirements of Specifications 3.2.2 and 3.2.3, or

b. Be in HOT STANDBY within 6 hours.

SURVEILLANCE REQUIREMENTS

4.10.2.1 The THERMAL POWER shall be determined to be less than or equal to 85% of RATED THERMAL POWER at least once per hour during PHYSICS TESTS.

4.10.2.2 The requirements of the below listed specifications shall be performed at least once per 12 hours during PHYSICS TESTS:

a. Specifications 4.2.2.2 and 4.2.2.3, and

b. Specification 4.2.3.2.

Justification and Safety Analysis

Specification 3.4.9.1.a (page 3/4 4-30) is proposed to be changed due to an inconsistency with Figure 3.4-2b. This figure is based upon a maximum heatup rate for Unit 2 of 60°F per hour. The change is clearly more restrictive and has no adverse safety significance.

Table 4.3-9, Note (2) (page 3/4 3-77) is proposed to be changed due to an administrative error. The current note (2) contains the word "and" after part c which implies that part d is missing. In fact, part d in the Standardized Technical Specification was intentionally deleted when the McGuire document was developed because it did not apply.

Two typographical errors in Table 3.7-5 (page 3/4 7-38) are proposed to be corrected as shown.

The ACTION section of Specification 3.7.12 (page 3/4 7-42) is proposed to be revised to properly reference the temperature limits shown in Table 3.7-6 and to be consistent with the LCO.

The proposed change to Specification 3.10.2 (page 3/4 10-2) involves deleting a reference to Specification 3.1.3.7 which does not exist.

Analysis of Significant Hazards Consideration

Because the proposed amendments involve only administrative or typographical changes, they do not involve significant hazards considerations according to the standards of 10 CFR 50.92.

Attachment 10

Proposed Amendments to McGuire Units 1 and 2
Technical Specifications Concerning Operation with an
Inoperable Channel of Steam Generator Level Instrumentation

TABLE 3.3-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
8. Overpower ΔT					
Four Loop Operation	4	2	3	1, 2	6#
Three Loop Operation	(**)	(**)	(**)	(**)	(**)
9. Pressurizer Pressure-Low	4	2	3	1	6#
10. Pressurizer Pressure--High	4	2	3	1, 2	6#
11. Pressurizer Water Level--High	3	2	2	1	7#
12. Low Reactor Coolant Flow					
a. Single Loop (Above P-8)	3/loop	2/loop in any operating loop	2/loop in each operating loop	1	7#
b. Two Loops (Above P-7 and below P-8)	3/loop	2/loop in two operating loops	2/loop each operating loop	1	7#
13. Steam Generator Water Level--Low-Low	4/stm. gen.	2/stm. gen. in any operating stm. gen.	3/stm. gen. each operating stm. gen.	1, 2	7#

Attachment 10, page 1

TABLE 3.3-1 (Continued)

ACTION STATEMENTS (Continued)

ACTION 3 - With the number of channels OPERABLE one less than the Minimum Channels OPERABLE requirement and with the THERMAL POWER level:

a. Below the P-6 (Intermediate Range Neutron Flux Interlock) Setpoint, restore the inoperable channel to OPERABLE status prior to increasing THERMAL POWER above the P-6 Setpoint, and

b. Above the P-6 (Intermediate Range Neutron Flux Interlock) Setpoint but below 10% of RATED THERMAL POWER, restore the inoperable channel to OPERABLE status prior to increasing THERMAL POWER above 10% of RATED THERMAL POWER.

ACTION 4 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement suspend all operations involving positive reactivity changes.

ACTION 5 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, verify compliance with the SHUTDOWN MARGIN requirements of Specification 3.1.1.1 or 3.1.1.2, as applicable, within 1 hour and at least once per 12 hours thereafter.

ACTION 6 - With the number of OPERABLE channels one less than the Total Number of Channels, STARTUP and/or POWER OPERATION may proceed provided the following conditions are satisfied:

a. The inoperable channel is placed in the tripped condition within 1 hour, and

b. The Minimum Channels OPERABLE requirement is met; however, the inoperable channel may be bypassed for up to 2 hours for surveillance testing of other channels per Specification 4.3.1.1, and Specification 4.3.2.1.

ACTION 7 - With the number of OPERABLE channels one less than the Total Number of Channels, STARTUP and/or POWER OPERATION may proceed until performance of the next required OPERATIONAL TEST provided the inoperable channel is placed in the tripped condition within 1 hour.

ACTION 8 - With less than the Minimum Number of Channels OPERABLE, within 1 hour determine by observation of the associated permissive annunciator window(s) that the interlock is in its required state for the existing plant condition, or apply Specification 3.0.3.

TABLE 3.3-3 (Continued)

ACTION STATEMENTS (Continued)

ACTION 18 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 48 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

ACTION 19 - With the number of OPERABLE channels one less than the Total Number of Channels, STARTUP and/or POWER OPERATION may proceed provided the following conditions are satisfied:

a. The inoperable channel is placed in the tripped condition within 1 hour, and

b. The Minimum Channels OPERABLE requirement is met; however, the inoperable channel may be bypassed for up to 2 hours for surveillance testing of other channels per Specification 4.3.2.1 and Specification 4.3.[illegible].

ACTION 20 - With less than the Minimum Number of Channels OPERABLE, within 1 hour determine by observation of the associated permissive annunciator window(s) that the interlock is in its required state for the existing plant condition, or apply Specification 3.0.3.

ACTION 21 - With the number of OPERABLE Channels one less than the Minimum Channels OPERABLE requirement, be in at least HOT STANDBY within 6 hours and in at least HOT SHUTDOWN within the following 6 hours; however, one channel may be bypassed for up to 2 hours for surveillance testing per Specification 4.3.2.1 provided the other channel is OPERABLE.

ACTION 22 - With the number of OPERABLE channels one less than the Total Number of Channels, restore the inoperable channel to OPERABLE status within 48 hours or be in at least HOT STANDBY within 6 hours and in at least HOT SHUTDOWN within the following 6 hours.

ACTION 23 - With the number of OPERABLE channels one less than the Total Number of Channels, restore the inoperable channel to OPERABLE status within 48 hours or declare the associated valve inoperable and take the action required by Specification 3.7.1.4.

ACTION 24 - With the number of OPERABLE channels less than the Total Number of Channels, restore the inoperable channel to OPERABLE status within 48 hours or declare the associated auxiliary feedwater pump inoperable and take the action required by Specification 3.7.1.2. With the channels associated with more than one auxiliary feedwater pump inoperable, immediately declare the associated auxiliary feedwater pumps inoperable and take the action required by Specification 3.7.1.2.

Justification and Safety Analysis

The proposed amendments would involve a change to Technical Specifications Table 3.3-1 concerning the action required in the event one of the four instrumentation channels per steam generator is inoperable which actuate reactor trip upon low-low steam generator water level. The proposed change would allow bypassing the inoperable channel for up to 2 hours for surveillance testing of the remaining operable channels.

The instrumentation channels which actuate reactor trip upon low-low steam generator level also actuate automatic start of the auxiliary feedwater pumps as required for Engineered Safety Features Actuation System (ESFAS) Instrumentation (shown in Table 3.3-3, item 7.c); however, the ACTION requirement for ESFAS Instrumentation for low-low steam generator level allows an inoperable channel to be bypassed for up to 2 hours for surveillance testing of other channels. Since the existing requirement for reactor trip instrumentation does not allow this, a forced outage will be required to perform the monthly surveillance testing if only one channel is inoperable. This is inconsistent with other requirements because (1) the "minimum channels operable" requirement is three per steam generator, and (2) the ESFAS Instrumentation requirement for this instrumentation allows bypassing the inoperable channel for surveillance testing, and (3) bypassing the inoperable channel is allowed for other reactor trip system functions (such as, low pressurizer pressure) having 2 out of 4 trip logic. Further, bypassing the inoperable channel for up to 2 hours is reasonable because it results in a temporary trip logic requiring 2 of 3 channels per steam generator to trip the reactor. Also, operation in this configuration would be infrequent because the only applicable surveillance testing is the Analog Channel Operational Test which is performed monthly.

The proposed changes to ACTION 6 on page 3/4 3-7 and ACTION 19 on page 3/4 3-24 would clarify the fact that some of the instrumentation perform dual functions - reactor trip and ESFAS.

Analysis of Significant Hazards Consideration

This analysis is provided in accordance with 10 CFR 50.91 and is performed according to the standards of 10 CFR 50.92.

The proposed amendments involve changes to achieve consistency throughout the Technical Specifications concerning the ACTION requirements for continued operation with one channel inoperable on reactor trip system instrumentation having 2 of 4 channels trip logic. Furthermore, the change would only affect operation under relatively unusual conditions and for a limited period of time - up to 2 hours during surveillance testing which is performed monthly. Therefore, the safety significance of the proposed changes is small.

The Commission has provided guidance concerning the application of standards of no significant hazards determination by providing certain examples (48 FR 14870). One of the examples of actions likely to involve no significant hazards considerations is a change to achieve consistency throughout the technical specifications. As described above, this example can be applied to the proposed amendments.

Based upon the above analysis, the proposed amendments are determined to involve no significant hazard consideration.

Attachment 11

Proposed Amendments to McGuire Units 1 and 2 Technical Specifications Concerning Verifying the Position of Inaccessible Sprinkler System Valves

LIMITING CONDITION FOR OPERATION (Continued)

APPLICABILITY: Whenever equipment protected by the Spray/Sprinkler System is required to be OPERABLE.

ACTION:

a. With one or more of the above required Spray and/pr Sprinkler Systems inoperable, within 1 hour establish a continuous fire watch with backup fire suppression equipment for those areas in which redundant systems or components could be damaged; for other areas, establish an hourly fire watch patrol.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.10.2 Each of the above required Spray and/or Sprinkler Systems shall be demonstrated OPERABLE:

a. At least once per 31 days, by verifying that each valve (manual, power-operated, or automatic) in the flow path which is accessible during plant operation is in its correct position,

b. At least once per 12 months, by cycling each testable valve in the flow path through at least one complete cycle of full travel,

c. At least once per 18 months:

   1) By performing a system functional test which includes simulated automatic actuation of the systems, and:

      a) Verifying that the automatic valves in the flow path actuate to their correct position on a Fire Detection test signal, and

      b) Cycling each valve in the flow path that is not testable during plant operation through at least one complete cycle of full travel.

   2) By a visual inspection of the dry pipe spray and sprinkler headers to verify their integrity; and

   3) By a visual inspection of each nozzle's spray area to verify the spray pattern is not obstructed.

   4) By verifying that each valve (manual, power-operated, or automatic) in the flow path which is inaccessible during plant operation is in its correct position,

d. At least once per 3 years, by performing an air flow test through each open head spray/sprinkler header and verifying each open head spray/sprinkler nozzle is unobstructed.

Justification and Safety Analysis

Specification 4.7.10.2.a currently requires verifying at least once per 31 days that sprinkler system valves are in their correct positions. The proposed amendments would make Specification 4.7.10.2.a not applicable to valves which are inaccessible during plant operation and would add Specification 4.7.10.2.c.4 to require verifying the positions of those valves at least once per 18 months. These changes would affect 9 valves per unit located in lower containment which are inaccessible during plant operation due to radiation levels.

Each of these valves are either locked in position or electrically supervised which is consistent with the guidelines in the Standard Review Plan. Because of this and because personnel entry into lower containment is controlled to a minimum, verifying the positions of inaccessible valves every 18 months is adequate to ensure the operability of the sprinkler systems.

Analysis of Significant Hazards Consideration

This analysis is provided as required by 10 CFR 50.91:

Because the valves affected by the proposed amendments are either locked or electrically supervised and because personnel entry into lower containment is controlled to a minimum, the probability of the valves being mispositioned is low. Therefore, extending the surveillance frequency from 31 days to 18 months will not significantly increase the probability or consequences of an accident previously evaluated.

The proposed amendments only affect surveillance frequencies and do not affect test methods, acceptance criteria, nor operating conditions. Therefore, new or different kinds of accidents are not created. Also, since no changes to accident analyses nor surveillance acceptance criteria are proposed, the proposed amendments will not affect a margin of safety.

Based on the above analysis according to the standards of 10 CFR 50.92, the proposed amendments do not involve significant hazards considerations.

The following valves would be affected by the proposed change to Specification 4.7.10.2.a.

Valve Numbers		
Unit 1	Unit 2	Valve Description
RF955	RF957	Pipe Corridor sprinkler system
RF843	RF844	Inline valve upstream of following valves & hoses
RF956	RF958	Control valve for 3 hoses in pipe corridor
RF881	RF885	Reactor Coolant pump C sprinkler system
RF985	RF987	Lower Cont. Filter B sprinkler system
RF880	RF884	Reactor Coolant pump B sprinker system
RF879	RF883	Reactor Coolant pump A sprinkler system
RF986	RF988	Lower Cont. Filter A sprinkler system
RF882	RF886	Reactor Coolant pump D sprinkler system

Attachment 12

Proposed Amendments to McGuire Units 1 and 2
Technical Specifications to Clarify Requirements
for the Primary Containment Distributed Ignition System

CONTAINMENT SYSTEMS

HYDROGEN CONTROL DISTRIBUTED IGNITION SYSTEM

LIMITING CONDITION FOR OPERATION

Both trains of the Hydrogen Mitigation

3.6.4.3 ~~The Primary Containment Distributed Ignition~~ System shall be OPERABLE.

APPLICABILITY: MODES 1 and 2.

ACTION:

Hydrogen Mitigation

With one train of the ~~Distributed Ignition~~ System inoperable, restore the inoperable system to OPERABLE status within 7 days or increase the surveillance interval of Specification 4.6.4.3.a from 92 days to 7 days on the OPERABLE train until the inoperable train is returned to OPERABLE status.

SURVEILLANCE REQUIREMENTS

Each train of the Hydrogen Mitigation

4.6.4.3 ~~The Distributed Ignition~~ System shall be demonstrated OPERABLE:

a. At least once per 92 days by energizing the supply breakers and verifying that at least ~~64~~ 32 of ~~66~~ 33 igniters are energized,* and

b. At least once per 18 months by verifying the temperature of each igniter is a minimum of 1700°F.

* Inoperable igniters must not be on corresponding redundant circuits which provide coverage for the same region.

Justification and Safety Analysis

The proposed amendments would revise the LCO and the Surveillance Requirements for the Primary Containment Distributed Ignition System to clarify that the system consists of two redundant trains. This is necessary to be consistent with the Action section.

Surveillance Requirement No. 4.6.4.3.a currently refers to 66 total igniters. This is actually two trains of 33 igniters each. Because the Action Section implies that one train can be considered operable while the other train is inoperable, operability must be defined on a "per train" basis. The proposed changes would result in a more restrictive definition of operability when two igniters on the same train are inoperable; this condition would be acceptable under the existing specification but not under the proposed specification.

Analysis of Significant Hazards Consideration

This analysis is provided as required by 10 CFR 50.91:

The Commission has provided guidance concerning the application of standards of no significant hazards determination by providing certain examples (48 FR 14870). One of the examples of actions likely to involve no significant hazards considerations is a change to achieve consistency throughout the technical specifications. As described above, this example can be applied to the proposed amendments.

Therefore, the proposed amendments do not involve significant hazards considerations.