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DUKE POWER

April 26, 1990

U.S. Nuclear Regulatory Commission
Document Control Desk
Washington, D.C. 20555

Subject: McGuire Nuclear Station Unit 2
Docket No. 50-370
Licensee Event Report 370/90-01-01

Gentlemen:

Pursuant to 10 CFR 50.73 Sections (a)(1) and (d), attached is Licensee Event Report 370/90-01-01 concerning an inoperable Lower Air Lock Control Valve for containment isolation. This report is being revised and submitted in accordance with 10 CFR 50.73(a)(2)(i). This event is considered to be of no significance with respect to the health and safety of the public.

Very truly yours,

Tony L. McConnell

T.L. McConnell

DVE/ADJ/cbl

Attachment

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MC-815-04
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NRC Form 366 (9-83)

U.S. NUCLEAR REGULATORY COMMISSION
APPROVED OMB NO. 3150-0104
EXPIRES 8/31/88

LICENSEE EVENT REPORT (LER)

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
McGuire Nuclear Station, Unit 2	0 5 0 0 0 3 7 0	1 OF 0 9

TITLE (4) A Containment Isolation Valve Was Inoperable Due To Unanticipated Interaction Of Components And Erroneous Documentation

EVENT DATE (5) MONTH	DAY	YEAR	LER NUMBER (6) YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	REPORT DATE (7) MONTH	DAY	YEAR	OTHER FACILITIES INVOLVED (8) FACILITY NAMES	DOCKET NUMBER(S)
0 1	2 2	9 0	9 0	— 0 0	0 1	0 3	1 4	9 0	N/A	0 5 0 0 0
										0 5 0 0 0

OPERATING MODE (9): 1
POWER LEVEL (10): 1 0 0

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more of the following) (11)

[ ] 20.402(b)	[ ] 20.405(c)	[ ] 50.73(a)(2)(iv)	[ ] 73.71(b)
[ ] 20.405(a)(1)(i)	[ ] 50.36(c)(1)	[ ] 50.73(a)(2)(v)	[ ] 73.71(c)
[ ] 20.405(a)(1)(ii)	[ ] 50.36(c)(2)	[ ] 50.73(a)(2)(vii)	[ ] OTHER (Specify in Abstract below and in Text, NRC Form 366A)
[ ] 20.405(a)(1)(iii)	[X] 50.73(a)(2)(i)	[ ] 50.73(a)(2)(viii)(A)	
[ ] 20.405(a)(1)(iv)	[ ] 50.73(a)(2)(ii)	[ ] 50.73(a)(2)(viii)(B)	
[ ] 20.405(a)(1)(v)	[ ] 50.73(a)(2)(iii)	[ ] 50.73(a)(2)(x)	

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER AREA CODE	
Alan Sipe, Chairman, McGuire Safety Review Group	7 0 4	8 7 5 - 4 1 8 3

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT (13)

CAUSE	SYSTEM	COMPONENT	MANUFAC-TURER	REPORTABLE TO NPRDS	CAUSE	SYSTEM	COMPONENT	MANUFAC-TURER	REPORTABLE TO NPRDS

SUPPLEMENTAL REPORT EXPECTED (14)

[ ] YES (If yes, complete EXPECTED SUBMISSION DATE) [X] NO

EXPECTED SUBMISSION DATE (15) MONTH DAY YEAR

ABSTRACT (Limit to 1400 spaces, i.e., approximately fifteen single-space typewritten lines) (16)

On January 23, 1990, Design Engineering personnel discovered while perfomring a field survey, that solenoid valve 2IASV5160, Lower Air Lock Control Valve, mounted on the Personnel Air Lock (PAL) bulkhead, was within 1/2 inch of the stationary Bypass Leakage Enclosure (BLE). Valve 2IASV5160 would be pinched between the PAL bulkhead and the BLE under a worst case Loss Of Coolant (LOCA) Design Basis Accident condition. On January 26, 1990, the lower PAL outside door was declared inoperable under Technical Specification (TS) 3.6.1.3 because of the potential inoperability of solenoid valve 2IASV5160. On February 1, 1990, the valve was relocated to a column within the annulus area near the PAL. On February 12, 1990, solenoid valve 2IASV5160 was retroactively determined inoperable under TS 3.6.2 as a containment isolation valve versus originally determined lower PAL outside door inoperability under TS 3.6.1.3. This event is assigned a cause of Unanticipated Interaction of Components because of field routing. The new valve bracket and replacement solenoid valve were relocated in the field by Construction Maintenance Division (CMD) personnel. A second cause of Erroneous Documentation was assigned because the variation notice associated with the field relocation did not reflect the actual location of the new valve bracket and solenoid valve. Design Engineering personnel determined the acceptability of the valve bracket and solenoid valve relocation based on the incorrect location information documented on the variation notice by CMD personnel. Unit 2 was in Mode 1 (Power Operation) at 100 percent power when this event was discovered. Unit 1 was not affected.

EVALUATION:

Background

The general purpose of the containment system is to provide a barrier confining potential releases of radioactivity from severe accidents. This is accomplished by maintaining leak tightness within specified bounds. As a design feature, the containment system is provided primarily for the protection of public health and safety. The free standing steel containment has an outer reinforced concrete Reactor Building and an annular space which is maintained at a lower-than-atmospheric pressure following a Loss Of Coolant Accident (LOCA). These structures form a double barrier to prevent the escape of fission products should a LOCA occur.

The containment isolation systems provide the means of isolating fluid systems that pass through Containment penetrations [EIIS:PEN] so as to confine to the containment any radioactivity that may be released following a design basis event. The containment isolation systems are required to function following a design basis event to isolate non-essential fluid systems penetrating the Containment.

The design bases for the containment isolation systems, in part, are indicated below:

A. A double barrier is provided for all fluid penetrations to assure that no single failure or malfunction of an active component can result in loss of isolation or intolerable leakage.

B. Upon receipt of either a Phase A Containment isolation signal which is derived from the safety injection signal or a Phase B Containment isolation signal which is derived from the high-high Containment pressure signal, the containment isolation system closes fluid line penetrations not required for Engineering Safety Features operation.

C. Isolation valves [EIIS:ISV] outside the containment are located as close to the Containment as practicable, and upon loss of actuating power, automatic isolation valves are designed to take the position that provides greater safety.

D. Isolation valving systems from a seismic consideration are designed the same as the piping system and/or the penetration of which they are a part, whichever is the higher classification.

Technical Specification (TS) 3.6.3 states, in part, that the containment isolation valves shall be operable in Mode 1 (Power Operation), Mode 2 (Startup), Mode 3 (Hot Standby), and Mode 4 (Hot Shutdown). With one or more containment isolation valve(s) inoperable, maintain at least one isolation valve operable in each affected penetration that is open and isolate each affected penetration within 4 hours by use of at least one closed manual valve or blind flange.

TS 3.6.1.3 states, in part, that each containment air lock [EIIS:AL] shall be operable with both doors [EIIS:DR] closed except when the air lock is being used for normal transit entry and exits through the containment, then at least one air lock door shall be closed in Modes 1 through Mode 4. With one containment air lock door inoperable, maintain at least the operable air lock door closed and either restore the inoperable air lock door within 24 hours or lock the operable air lock door closed.

Solenoid valve 2IASV5160, Lower Personnel Air Lock Control, functions as an outside containment isolation valve, Figure 1 page 8. Valve 2IASV5160 was mounted on the Auxiliary Building [EIIS:NF] side of the personnel air lock (PAL) barrel bulkhead. Displacement of the air lock bulkhead under postulated LOCA and seismic conditions are calculated in accordance with MCS 1108.00-00-0002, Specification For The Response Spectra And Seismic Displacements For Category 1 Structures, Revision 5. These displacements are 1.36 inches radially outward, 0.39 inches vertically and 0.21 inches laterally.

NUREG-0660, NRC Action Plan Developed as a Result Of The TMI-2 Accident, item I.D.2 required licensees to design and install a Safety Parameter Display System (SPDS). The SPDS was implemented on Unit 1 and Unit 2 in November, 1984. The NRC Staff and its Consultants made recommended changes based on an onsite design verification/validation audit of the SPDS performed in May, 1985. Duke Power Company agreed to make the required changes to the SPDS as outlined in a February, 1988, response, which included adding a containment isolation verification logic.

Description of Event

In September 1989, Nuclear Station Modification (NSM) MG-22138 was implemented. Non-position indicating solenoid valves were replaced with position indicating solenoid valves to provide inputs to the SPDS.

On January 23, 1990, Design Engineer A performed a field survey of the routing of non-safety related air supply tubing to the safety related solenoid valves 2IASV5080, Upper Personnel Air Lock Control and 2IASV5160. This field survey was performed in conjunction with a review of NSM MG-12138 for Unit 1 which was similar to implemented NSM-MG-22138. Design Engineer A discovered that portions of valve 2IASV5160 were within 1/2 inch of the stationary Bypass Leakage Enclosure (BLE). This condition could result in an interference during a LOCA.

On January 26, 1990, at approximately 1000, the Lower PAL outside door was declared inoperable under TS 3.6.1.3 because of the potential inoperability of valve 2IASV5160. Appropriate TS action was complied with, in that the inside PAL door was secured.

On February 1, 1990, valve 2IASV5160 was relocated to an Auxiliary Building column within the annulus area near the PAL under exempt variation notice MEVN-2185.

On February 12, 1990, valve 2IASV5160 was retroactively determined inoperable under TS 3.6.3 versus the Lower PAL outside door being inoperable under TS 3.6.1.3. Under the postulated worst case design basis accident conditions of a LOCA, valve

2IASV5160 would have been forced against the BLE. If the valve had not closed prior to being pinched between the PAL bulkhead and BLE, there would be no assurance that the valve could be closed or that the valve would maintain a tight seal. Also, no credit could be assumed for the inside PAL door being closed.

Conclusion

This event is assigned a cause of Unanticipated Interaction of Components because of field routing. Design Engineer B reviewed the original modification for the acceptability of the valve bracket location. The design review was performed using manufacturer drawings of the valves and PAL. Design Engineer B concluded that the replacement solenoid valve would fit at the existing location. This conclusion was based on the new valve having the same mounting bolt pattern and being approximately 7 inches taller. This design review was performed in September, 1988, based on the best recollection of Design Engineer B. In July 1989, Design Engineer B was informed that there was interference preventing the new valve installation. Design Engineer B discovered that the new valve was approximately 5 1/2 inches taller than anticipated. Design Engineer B performed a design determination on July 24, 1989, for variation notice MC-2109 allowing the PAL barrel lip to be notched to provide clearance for the new valve. On August 17 or 18, 1989, Projects Services personnel discussed the possibility of relocating the valve with Design Engineer B. Three possibilities were discussed. The valve could be relocated to the side of the PAL barrel. An alternate location was on an Auxiliary Building column, which would facilitate maintenance. The third proposal was to enlarge the access penetration through the BLE. Relocation to the PAL barrel would have required up to four days of design analysis. The relocation to the Auxiliary Building column would have required up to two weeks of design analysis with no guarantee that the results would have been acceptable. Enlarging the access penetration was the acceptable solution chosen by Project Services personnel since this choice would impact the schedule the least. The BLE access penetration was enlarged under variation notices MC-2221 and MC-2222. Construction Maintenance Division (CMD) personnel field routed the new valve within the enlarged BLE penetration to facilitate installation and maintenance. The field routing of the valve location was documented on variation notice MC-2224. Design Engineer B performed a design review on the valve relocation based on information documented on variation notice MC-2224.

A second cause of Erroneous Documentation was assigned because variation notice MC-2224 did not reflect the actual location of the relocated valve bracket. Design Engineering personnel determined that the new location of the valve bracket/solenoid valve was acceptable based on the incorrect location information documented in the variation notice. The variation notice reflected the use of an extended bracket and a new elevation of approximately 730 feet 0 inches. The variation notice did not reflect any lateral relocation from the drawing specified 4 foot 3 inches PAL barrel center line to valve bracket center line. Design Engineer B took exception to the new bracket elevation since it represented a drop of over 12 feet. CMD personnel clarified the drop to be approximately 1 foot 2 inches versus 12 feet. Field surveys performed by Design Engineer B and CMD personnel on February 23 and 26, 1990, verified the actual bracket location. The new bracket was located approximately 1 3/4 inches laterally toward the outside of

*U.S. GPO: 1988-520-589 000

the PAL barrel. Also, the bracket elevation was approximately 741 feet 10 inches versus approximately 741 feet 4 inches. Figure 2 Page 9 depicts graphic orientation of the different bracket locations. It could not be determined during this investigation how the incorrect location data was derived.

A review of the Operational Experience Program Data Base for the previous twelve months revealed no events involving Containment Isolation valves that was attributed to Unanticipated Interaction of Components because of field routing or Erroneous Documentation.

This event is not Nuclear Plant Reliability Data System (NPRDS) reportable because the solenoid valve did not actually fail.

There were no personnel injuries, radiation overexposures, or uncontrolled release of radioactive material as a result of this event.

CORRECTIVE ACTIONS:

Immediate: The inside PAL door was secured in the closed position to provide an additional flow path barrier from the Containment Building.

Subsequent: Solenoid valve 2IASV5160 was relocated to an Auxiliary Building column under variation notice MEVN-2185. This was implemented by 2/1/90.

Planned:

1) CMD-North Engineering and Craft personnel involved with the field relocation of valve 2IASV5160 will be instructed to ensure that all variations from original design are communicated and documented accurately.

2) Project Services personnel will ensure that proper engineering analysis is performed, including Design Engineering review, and relocation coordinates are clearly documented when filed relocation of components is necessary.

3) The Safety Review Group will perform an investigation to determine if any generic problems exist with the field routing practice.

SAFETY ANALYSIS:

The installation discrepancy involving the PAL solenoid valve 2IASV5160 is a reportable condition because it is possible that the ability to maintain containment integrity could have been compromised due to the inoperability of the solenoid valve. The valve serves as the outside containment isolation for the bulkhead air tubing penetration. Check valve 2IACV5350, Lower Personnel Air Lock Control, provides for inside isolation. The solenoid valve, check valve, and connecting tubing [EIIS:TBG] are QA Condition 1, Quality Assurance Program applied to nuclear safety-related activities in accordance with 10CFR50 Appendix B. However, a portion of tubing circuitry between the inside check valve and the door seal [EIIS:SEAL] air tanks [EIIS:TK] including a section of flexible hose, is non-QA Condition 1.

A LOCA and seismic event can cause outward movement of the bulkhead a distance of 1.36 inches, with the solenoid valve becoming pinched, resulting in valve body distortion after bulkhead movement of 0.5 inches. Inoperability of the solenoid valve would then be due to a postulated inability of the valve to close under the distorted condition to accomplish the required Containment Isolation function. The details of the relationship involving bulkhead movement due to increasing containment pressure with time following the LOCA event are unknown for the purpose of this investigation. Since the Phase A Hi Containment Pressure signal occurs at a pressure of 1.0 psig during the initial seconds of the accident, it is proposed that the solenoid valve would have fulfilled its safety function before valve body distortion takes place. There is no assurance that the valve would have maintained a tight seal after valve body distortion occurred. It is proposed that any leakage would have been insignificant because of the valve body design. The valve body is machined from stainless steel plate and measures approximately 3x4x4 inches. The bonnet, disc guide, and disc are machined from stainless steel bar stock. The bonnet retains and aligns the disc guide which in turn aligns the disc. The bonnet is threaded and seal welded to the valve body. Valve body to disc seal is accomplished by an elastic O-ring. A combination of the valve body ruggedness and forgiving elastic sealing surface should have precluded significant leak by. Also, no valve distortions have been noted during Integrated Leak Rate Tests.

A postulated breach of containment integrity would require the simultaneous occurrence or existence of all of the following:

A) the inside PAL door is open or both of the associated inflatable door seals fail,

B) a break in the non-QA Condition 1 section of tubing has occurred,

C) the inside containment isolation check valve fails to isolate reverse flow, and

D) the outside containment isolation solenoid valve fails to close.

Since this safety analysis is based on the premise that a LOCA or seismic event has caused the predicted displacement of the bulkhead, also it is reasonable to assume that items B) and D) above would also have occurred. Protection against containment leakage is provided by check valve 2IACV5350, the failure of which is addressed in item C). This protection would only be necessary in the event that containment pressure (assume 15 psig) became greater than that existing in the Instrument Air System supply line as would be a case in a local depressurization of the Instrument Air piping. This would present a containment leakage path to either the annulus area or the auxiliary building atmosphere, where the leakage would be treated by the respective ventilation system. However, the likely leakage flowpath would be into containment due to the presence of normal Instrument Air System pressure of approximately 100 psig.

Concern for items B), C), and D) is of significance only in the event that item A), the door breach, has occurred. As previously noted, the installation discrepancy is known to have existed since the unit's previous refueling outage (mid-1989). With a return to service following the outage, containment entry through the PALs

would have been permitted only under unusual circumstances and controlled by strict administrative measures. It is a reasonable assertion that during Modes 1 through 4, when Containment Integrity is to be maintained, the number of PAL door openings would have been minimal and that any possible individual openings would have been of such short duration as to be negligible in consideration of the accident scenario. An unattended opening of the PAL door is unforseen, but would be detected by a Control Room [EIIS:NA] (CR) alarm [EIIS:ALM]. Therefore, the probability of occurrence of that described by item A) is insignificant.

For the purpose of this safety analysis, it is not necessary to consider door seal failure as part of the accident scenario. Two redundant seals (for testing purposes) are supplied with air from two independent air tanks, equipped with respective check valves for air loss protection. The air supply tanks and tubing are QA Condition 1. Pressure switches [EIIS:PS] are provided for CR indication of inflatable seal status.

In conclusion, although the lower PAL solenoid valve 2IASV5160 was determined to be technically inoperable due to an installation discrepancy, the coincident conditions necessary for an actual breach of containment integrity are unlikely. A containment barrier backup is also available for this specific situation in the form of the Annulus Ventilation System [EIIS:VD].

This event did not affect the health and safety of the public.

NRC Form 366A (9-83)

U.S. NUCLEAR REGULATORY COMMISSION

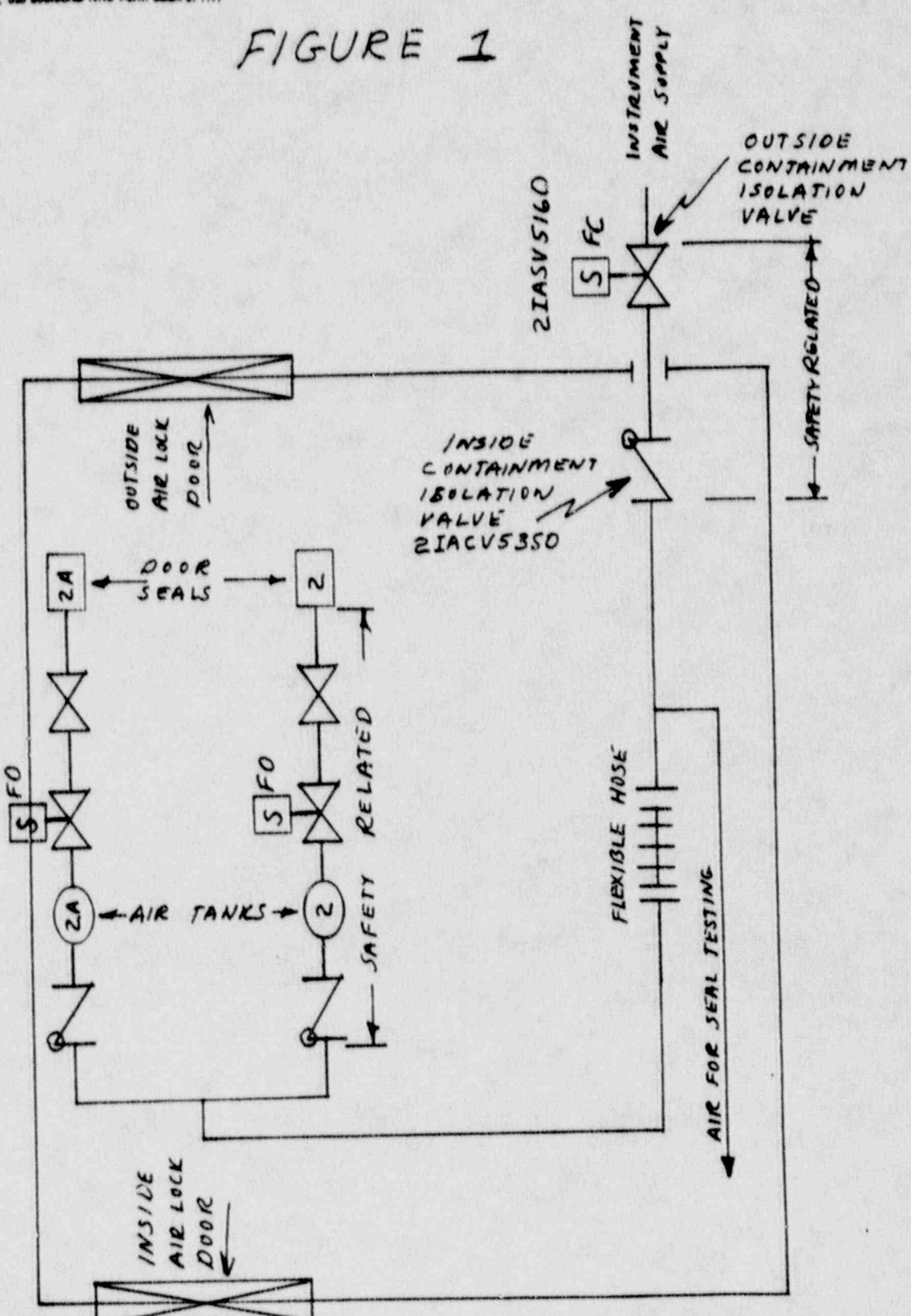
LICENSEE EVENT REPORT (LER) TEXT CONTINUATION

APPROVED OMB NO. 3150-0104
EXPIRES: 8/31/88

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6) YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	PAGE (3)
McGuire Nuclear Station, Unit 2	0 5 0 0 0 3 7 0	9 0	— 0 0 1	— 0 1	0 8 OF 0 9

TEXT (If more space is required, use additional NRC Form 366A's) (17)

FIGURE 1

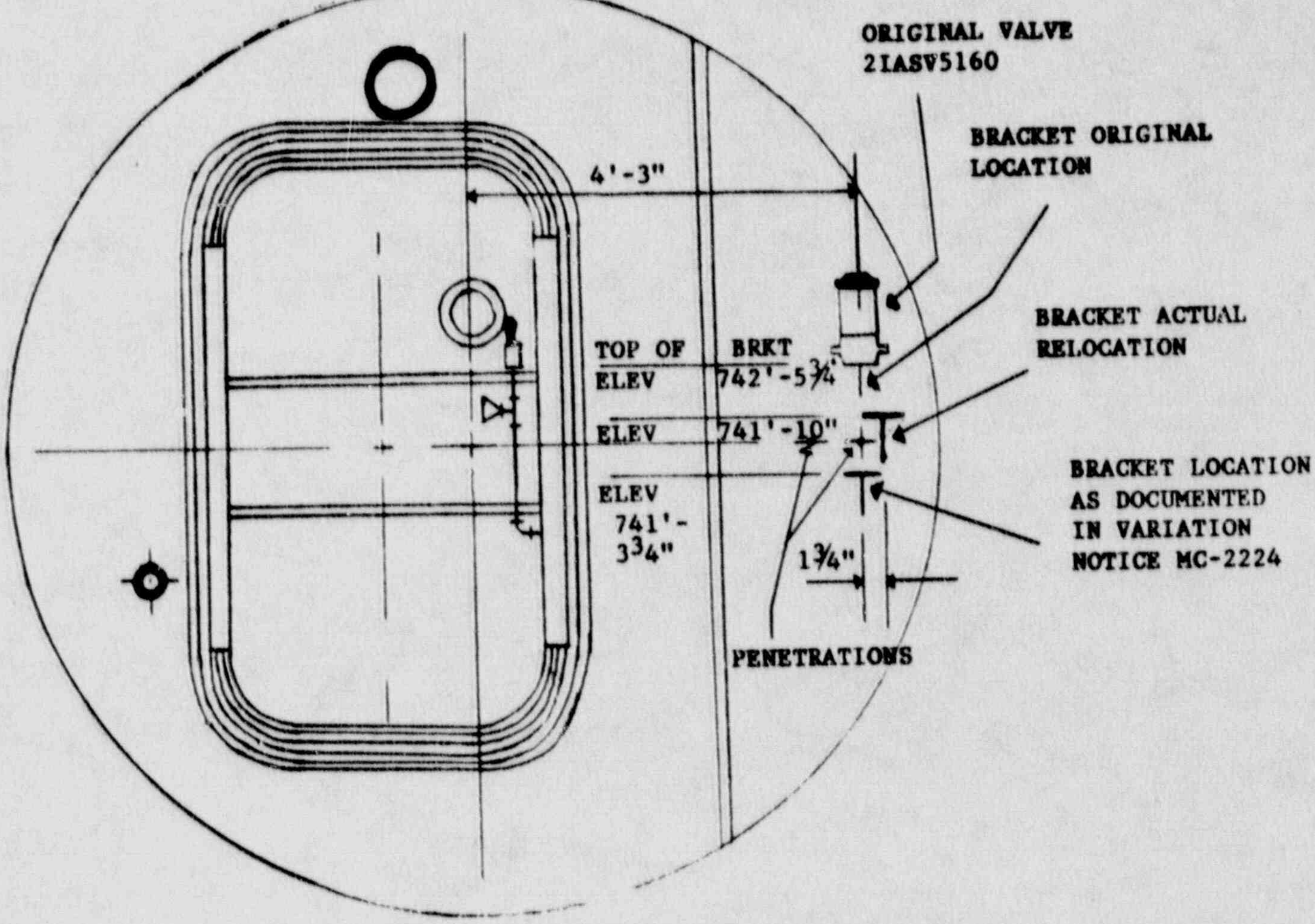


FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6) YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	PAGE (3)
McGuire Nuclear Station, Unit 2	0 5 0 0 0 3 7 0	9 0	— 0 0 1 —	0 1	0 9 OF 0 9

TEXT (If more space is required, use additional NRC Form 366A's) (17)

FIGURE 2

VALVE MOUNTING BRACKET
LOCATION DEPICTION



PAL BULKHEAD
AUX. BLDG. SIDE

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6) YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	PAGE (3)
McGuire Nuclear Station, Unit 2	0 5 0 0 0 3 7 0	9 0	0 0 1	0 1	0 1 OF 0 2

TEXT (If more space is required, use additional NRC Form 366A's) (17)

ADDITIONAL INFORMATION:

The following information addresses Planned Corrective Action number 3, to submit information on the results of the investigation erformed by McGuire Safety Review Group (MSRG) personnel and Quality Assurance Department personnel on field routing practices used by Construction And Maintenance Department (CMD) personnel when implementing Nuclear Station Modifications (NSMs).

This investigation consisted of interviews with various CMD Support personnel, CMD Craft personnel, Nuclear Production Department (NPD) Project Services personnel, and Design Engineering (DE) personnel involved in the NSM process at McGuire Nuclear Station (MNS).

The scope of these interviews covered all phases of the NSM process. The knowledge and expertise exhibited by the CMD Support personnel interviewed was noteworthy. They are experienced, conscientious, and very much involved throughout the NSM process. Generally, all personnel interviewed were consistent in their description of the process used in preparation and implementation of NSMs.

It is felt that every effort is made to install modifications as specified by Design drawings; however, it was noted that tolerances are also specified in Design Specifications and CMD personnel use these tolerances when necessary to avoid interferences. Design Specification tolerances are not used if tolerances are specified otherwise on Design Drawings for the NSM.

It was also noted that Instrument Control Specification ICS-A-11, Instrument Standards Installation Field Practices, specifies the following tolerances:

2.2.5 On a device located by Design Engineering, every effort shall be made to mount that device precisely where specified. If this cannot be accomplished, then the device can be mounted up to two feet in any direction from specified location.

A.5.3 Safety related instruments which are functionally redundant must meet minimum physical separation requirements. Minimum horizontal separation of functionally redundant safety related instruments is 18 inches.

This specification could allow a component to be moved a considerable distance without notifying DE personnel. This could result in components being mounted without interaction evaluations being performed.

Further investigation revealed that a new specification MC-1381.02-00, Instrumentation and Controls Field Installation Specification, was released by DE on March 8, 1990. The intent of the specification MC-1381.02-00 is to act as a source that references all I&C related documents and to provide a clearer, more concise, and more consistent specification overlap with documents such as ICS-A-11, etc. When the approved specification is issued in September, 1990, the old documents that overlap with the specification will be maintained only for historical purposes, and the new specification will be the governing document.

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6) YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	PAGE (3)
McGuire Nuclear Station, Unit 2	0 5 0 0 0 3 7 0	9 0	— 0 0 1	— 0 1	0 2 OF 0 2

TEXT (If more space is required, use additional NRC Form 366A's) (17)

Comments are to be provided to DE by May 14, 1990. CMD Management personnel have committed to provide training on the new specification to all appropriate CMD personnel by December 31, 1990.

When contacted about this potential problem, DE personnel indicated that they had no problem with the tolerances as specified; however, they agreed to note in their review of the I&C Study Draft Specification that these tolerances do not apply when tolerances are specified on the drawing. DE personnel also indicated that they would also consider changing the new specification to add limitations on field routing to certain areas or to list boundaries where these tolerances would not be applicable. It was also noted that tolerances specified in ICS-A-11 were changed in the new specification as follows:

7.17.3 On a device located by Design Engineering, every effort shall be made to mount that device where specified. The field may deviate from specified location up to two (2) feet in horizontal and three (3) inches in vertical direction. Deviations beyond there limits require prior approval by Design Engineering. All other installation requirements must be met.

This new specification should significantly reduce the possibility of components being be mounted without an evaluation by DE.

ADDITIONAL CORRECTIVE ACTION:

CMD Management personnel will provide training on specificaiton MC-1381.02-00, Instrumentation And Controls Field Installation Specification, to all appropriate CMD personnel.

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