ATTACHMENT 1

LOOSE THERMAL SLEEVE

SAFETY EVALUATION

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LOOSE THERMAL SLEEVE
SAFETY EVALUATION

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1.0 SUMMARY

Extensive evaluations were performed to determine the effects of loose reactor coolant pipe thermal sleeves at the Duke Power Company McGuire Unit 1 plant. These evaluations assumed all of these thermal sleeves become loose and are transported in the reactor coolant system as single units or fragments. This evaluation has concluded that reasonable assurances exist that safe plant operation is not compromised. Operation of the plant without thermal sleeves is also acceptable.

2.0 INTRODUCTION

2.1 PURPOSE

Westinghouse was recently informed by one of its operating plant customers that an underwater television inspection had revealed a loose metal piece under the reactor internal lower core plate. Subsequent investigations by Westinghouse and the utility resulted in the discovery of additional loose parts in the reactor vessel and an eventual conclusion that the sources of the parts were the thermal sleeves from the 10 inch RHR/SIS line nozzles. That conclusion has been verified by radiographic examination of all four such nozzles on the affected unit. The sleeves traveled through the cold leg into the reactor vessel where all missing parts have been accounted for and recovered. Radiographic examination of other similarly designed sleeves on the affected unit have revealed one broken weld and a very slight movement of the 14 inch surge line nozzle sleeve as well as an indication of a possible crack of a thermal sleeve weld in one of the two 3 inch charging lines.

On McGuire Unit 1 a radiographic examination (confirmed by visual) has indicated that one of the 10 inch RHR/SIS line nozzle thermal sleeves is missing. Loose parts monitoring indicates that this sleeves has migrated to the bottom of the reactor vessel.

As a result of the discovery of failed thermal sleeves, a safety evaluation was performed on the effects of loose and missing reactor coolant pipe nozzle thermal sleeves. This report summarizes and documents that safety evaluation.

2.2 THERMAL SLEEVE INVENTORY

Thermal sleeves are utilized in several locations in the McGuire plant Reactor Coolant System (RCS) to reduce thermal stresses on RCS pipe nozzles. The following list provides locations, sizes, and number of the reactor coolant pipe thermal sleeves. The pressurizer and reactor vessel upper head thermal sleeves are not presented since they are not considered in this evaluation, for the reasons as described in Section 4.7.

Location	Number	Length	I.D.	O.D.	Thickness	Weight
14" Pressurizer Surge Line (Hot Leg)	1	17.75	10.9"	11.25"	3/16"	30.5 lbs
10" SI/Accumulator Injection Line (Cold Leg)	4	14.6"	8.33"	8.625"	.149"	17.5 lbs
3" Charging Injection Lines (Cold Leg)	2	6.75"	2.12"	2.5"	3/16"	2.75 lbs

The material of construction of the thermal sleeves is A240 or A312 stainless steel, type 304 or 316.

2.3 ASSUMPTIONS

To complete the safety evaluation for McGuire Unit 1 certain assumptions were made. These assumptions are based on known facts, information gathered from the first operating plant to discover missing thermal sleeves, engineering judgement, and recommended actions for continued operation. The assumptions are as follows:

1. All reactor coolant piping thermal sleeves are assumed to come loose and are transported through the RCS system. Examinations confirm that that one 10 inch sleeve is not in place. Based on recent monitoring this sleeve is assumed to be in the reactor vessel lower plenum.

2. The sleeves are assumed to remain intact or split into quarter sections, whichever case provides the most conservative evaluation. The sleeves are attached by two welds at 180° in line with the loop flow on the upstream end. Field examinations indicates cracking

can occur at the welds allowing an intact sleeve to come loose. Another failure mode which has been observed is cracking of the sleeve along its length, beginning at one of the notches along the upstream end of the sleeve. Both of these failure modes produce large objects. The ductile nature of the sleeve material also makes it unlikely that small pieces would be generated by impacts within the reactor coolant system. This evaluation specifically considered objects ranging in size from a complete 14 inch sleeve to one quarter sections of the 3 inch sleeves. Smaller fragments were also addressed in the nuclear fuel evaluation. By considering quarter sections of the 3 inch thermal sleeve a range of fragments has been covered to complete a safety evaluation.

3. The plant operators are aware of the potential for loose parts and will monitor plant operations, pertinent equipment characteristics, and specifically loose parts monitoring system.

4. Inspections will be conducted to assess the condition of thermal sleeves that remain in place.

3.0 NOZZLE INTEGRITY

3.1 INTRODUCTION

This section summarizes the stress evaluation of the 3" charging nozzles, the 10" accumulator nozzles, and the 14" pressurizer surge nozzle on the main reactor coolant loop piping, to insure the structural integrity of the said nozzles assuming certain failures of the thermal sleeves.

The postulated thermal sleeve failure was assumed to be similar in nature to that found in the worst case of the 10" accumulator nozzles as described in Section 2.1 of this report.

The analysis included an evaluation of the subject nozzles without a thermal sleeve and a "bounding" evaluation of the nozzle at the location of the failed sleeve/nozzle attachment tack weld. This evaluation which considered all design transients and mechanical loads specified in the piping design specification demonstrates the structural integrity of the subject nozzles without thermal sleeves.

Due to the similarities in the geometry of all three nozzles, and the similarities in the thermal sleeve designs (see Figure 3.1) the same analytical techniques to be applied to all three nozzles. The evaluation was separated into the following three basic regions on the nozzle, (see Figure 3.1), 1) the location of the nozzle to pipe field weld at the "safe-end" of the nozzle, 2) the location of the original sleeve tack weld to nozzle wall and 3) the remaining body of the nozzle including the crotch region.

3.2 STRESS ANALYSIS

The stress analysis performed on the subject nozzles can be summairzed as follows. The detailed geometry and material of the nozzle, without a thermal sleeve, was obtained from the appropriate specifications. (For example, the previously mentioned figure and the plant specific drawings and equipment specifications). Then a detailed 2-dimensional finite element model was developed for the nozzle and appropriate representative portions of the large header pipe and attached branch pipe (Figures 3.2 and 3.3).

Using piping design specifications containing operating transient descriptions developed on the basis of the systems design criteria, the temperature transients, fluid velocities, number of occurrences, etc. were summarized for all applicable transients, and appropriate loading conditions were developed for the heat transfer analysis using the finite element model. The analysis included a time-history thermal loading for a sufficient duration of time to insure the maximum stress intensities were calculated for all locations.

Using the same finite element model, stress intensities were calculated from the pipe wall temperature distribution obtained from the heat transfer analysis for all critical locations. The actual fatigue evaluation of the component incorporates the methods and guidelines specified in the ASME Code Section III, Subsection NB, and all applicable Appendices.

This rigorous treatment has been applied to the 3" charging nozzle and the 14" surge nozzle without thermal sleeves. Due to design modifications for later plants, the 90°-10" accumulator nozzle was changed to a 45° inclined injection nozzle without a thermal sleeve. A complete set of thermal transient stress analysis was performed for this inclined injection nozzle for the same loading conditions as specified for the 90° injection nozzle. In addition, analysis was also performed on a geometrically similar nozzle (6-inch) without a thermal sleeve with similar design transients. The results of these two analysis were used in the qualification of the 10-inch accumulator injection nozzle without a thermal sleeve.

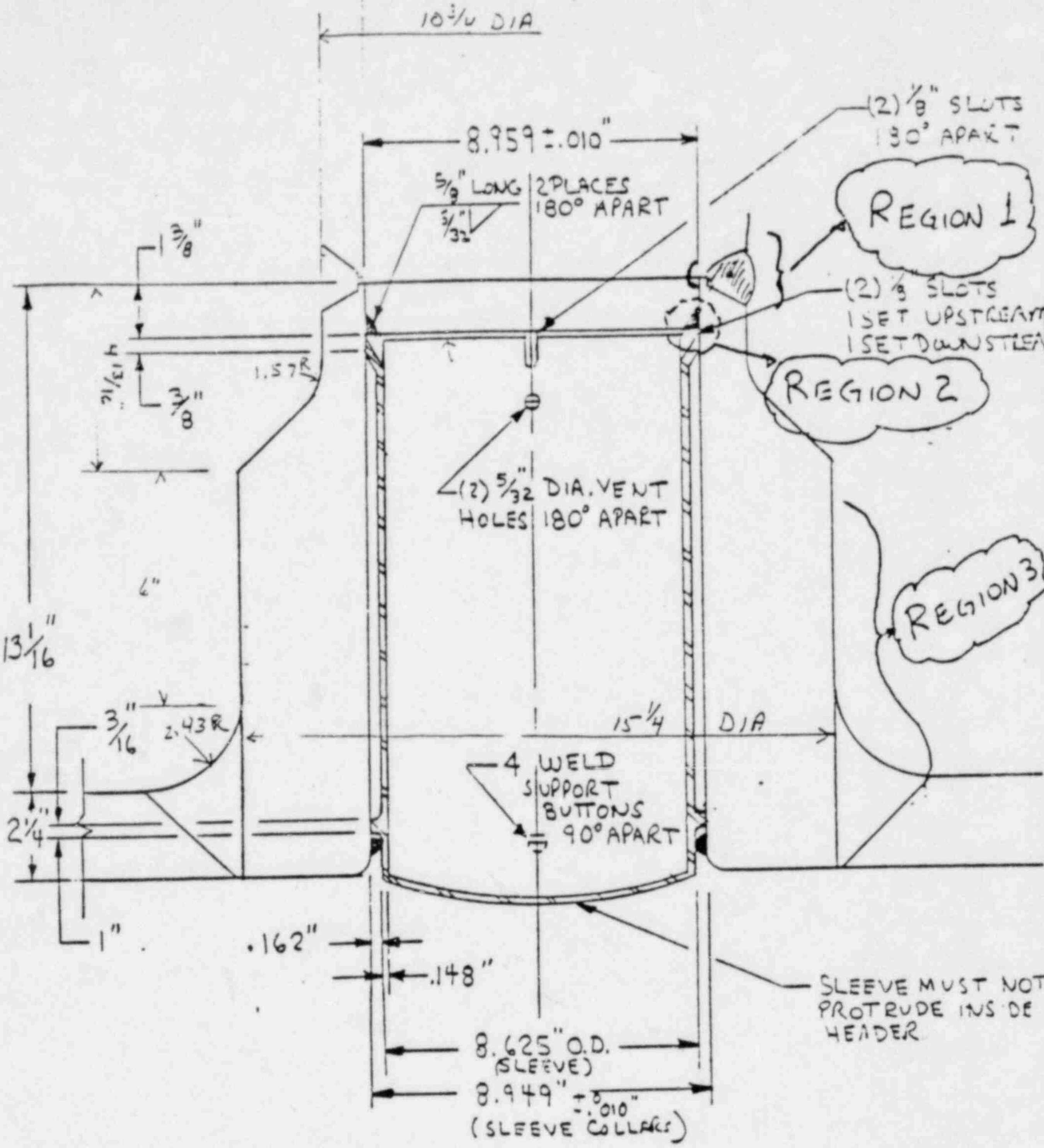
In the analysis of the nozzle without thermal sleeves, two locations were found where maximum peak stress intensity and fatigue usage occurred, 1) the thick part of the nozzle near the crotch region and 2) the nozzle to the branch pipe field weld. This second region was found to be critical after stress intensification factors were applied to the weld location, as specified in the ASME Code. Assuming the as-welded conditions, a stress concentration factor of 1.7 was applied on top of the calculated values. At the crotch region, a factor of only 1.1. was applied, due to the ground flush condition at the weld at that location.

The complete the fatigue calculation, the external loadings on the nozzle, as calculated for the McGuire Unit 1 plant were incorporated, and a usage factor calculated for each nozzle.

Finally, an evaluation of the failed tack weld region on the nozzle was performed. Because of the close proximity of the tack weld location to the pipe/nozzle butt weld (1.0-1.5 inches), the evaluation of the safe-end location could be shown to yield the same usage factor, once the following was considered. First, an appropriate stress intensification factor was required to simulate the inside surface of the nozzle at this location. Factors of 1.4 for K_3 and 1.5 for K_2 were conservatively used. This was based upon the relative severity of the conditions which resulted in the factors (K_3=1.7 and K_2= 1.8,) for an as-welded butt weld, (i.e., affected inside surface, thin-walled pipe, misalignment of the butted pipe walls,) and the condition actually present at the tack weld location (affected inside surface, thick wall pipe, perfect alignment). This difference in stress intensification factors more than offset the small increase in stress intensity due to the location being closer to the thick part of the nozzle and resulted in no significant change in stress.

3.3. CONCLUSIONS

The cumulative usage factors calculated on the basis described in the previous sections and the external loadings based on Duke/McGuire plant specific as-built information indicates that all critical locations meet the ASME Code requirements. Therefore, it is concluded that the nozzles are qualified to withstand all applicable design transients and will maintain their structural integrity without thermal sleeves for the plant design life.



10" THERMAL SLEEVE ON 27 1/2" HEADER

THERMAL SLEEVE APPROX. LENGTH = 13 15/16"

FIGURE 3.1

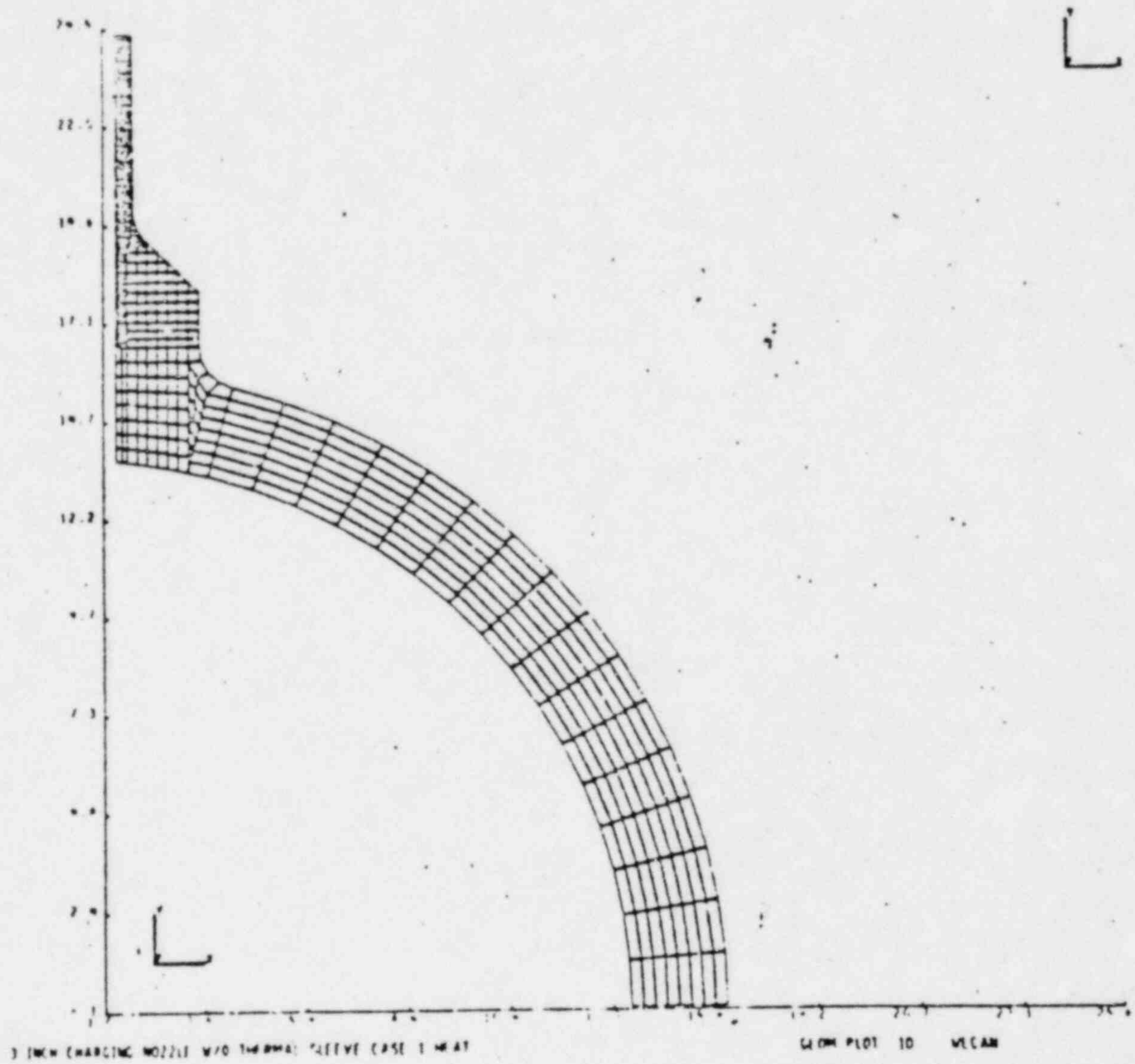

3 INCH CHARGING NOZZLE W/O THERMAL SLEEVE CASE 1 HEAT
GEOM PLOT 10 WECAN

FIGURE 3.2

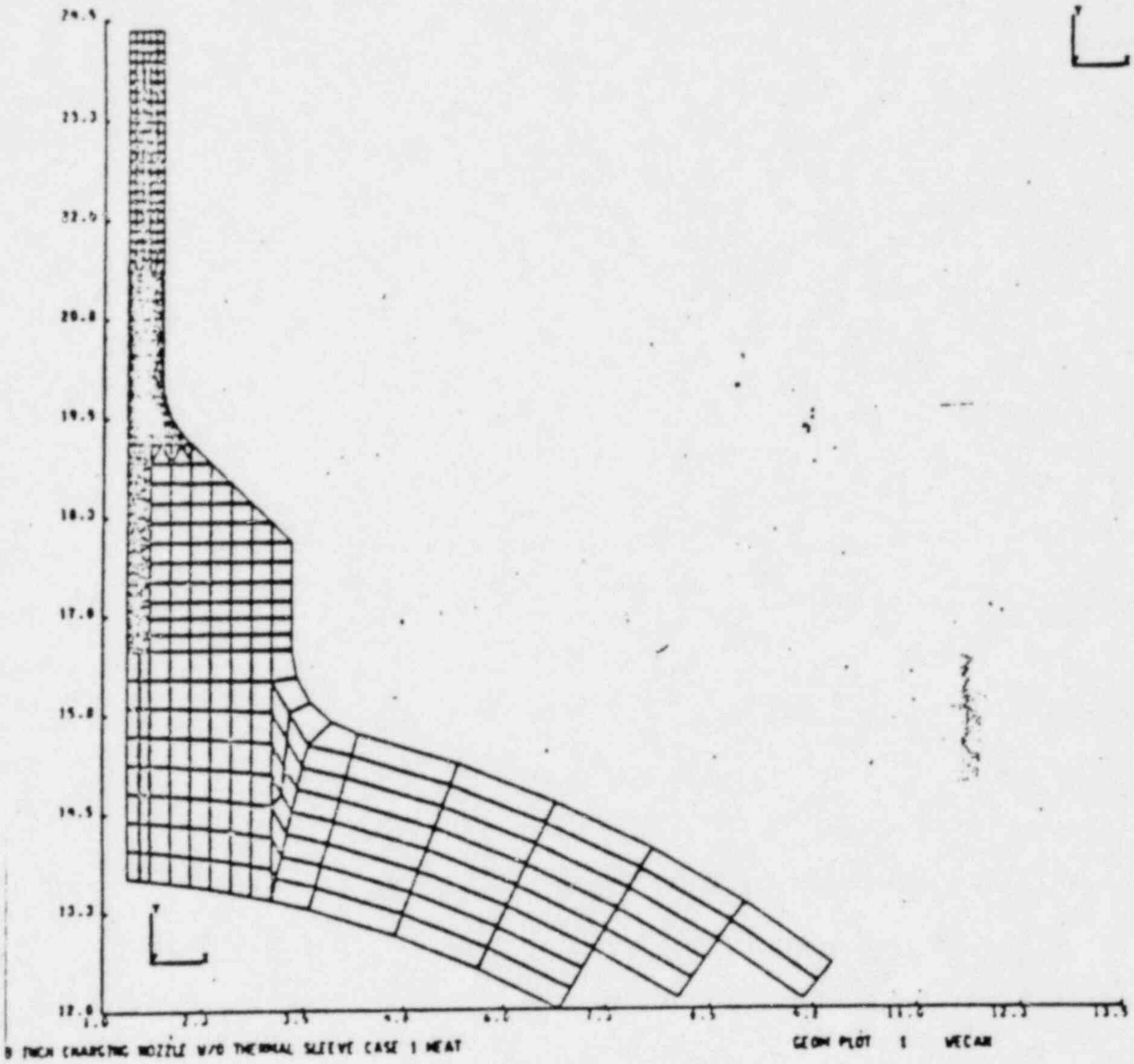

INCH CHARGING NOZZLE W/O THERMAL SLEEVE CASE 1 HEAT
GEOM PLOT 1 WECAN

FIGURE 3.3

4.0 MECHANICAL EFFECTS OF LOOSE OBJECTS

4.1 REACTOR COOLANT PIPE

The effect of the loose thermal sleeves on the primary system piping, either through impact or erosion, is expected to be negligible due to the limited impact energy created by the low radial flow velocities in the piping. The ductile material of the piping and the thermal sleeve would also preclude any sharp impact marks on the piping, thus eliminating any concern regarding possible stress concentration points.

The locations of the RTD bypass scoops and thermowells in the piping are all upstream of thermal sleeve locations, thus precluding damage by failure of a sleeve during operation. Reverse flow in the loop piping during plant startup could cause impacts on the scoops or thermowells, however, any damage causing pressure boundary failure or loss of an instrument would be detectable by the operator prior to criticality.

4.2 STEAM GENERATOR

4.2.1 Introduction

This evaluation considers the potential effects of an 11.5" outside diameter thermal sleeve entering the primary side of the Model D2 steam generators at McGuire Unit 1 from the connection of the pressurizer surge line. Affected components of the steam generator may include the tube sheet, divider plate, channel head, tube-to-tubesheet welds, tubesheet-to-divider plate weld, and the divider plate-to-channel head weld. Damage as a result of the 30.5 pound sleeve impacting on these components is considered separately in the following sections.

4.2.2 Tube Sheet and Tubes

The tubesheet of Model D2 steam generators is clad with Inconel 600 which is quite ductile. Repeated impacting on the cladding by the 304 stainless steel thermal sleeve, which is also a ductile material, would not be expected to cause the cladding to crack and break loose. Additionally, this impacting of ductile materials would tend to minimize the potential for the formation of any sharp edge.

The design of the D2 steam generators incorporates tube ends that are slightly recessed from the primary face of the tubesheet. No damage to the tube ends is expected since they would not be exposed directly to the impacting. Also, the potential for formation of sharp edges that may be able to fit inside the tube opening is low.

The thermal sleeve design contains notches at the upper ends for stress relief. These notches are 90° apart and experience indicates some cracking at these notches on failed sleeves. Thus, if the sleeve were to break apart it is anticipated to break at the notches, forming large sections. No piece that would be small enough to fit into the .664" inside diameter opening of the tube is expected to be formed from the break-up of the thermal sleeve.

Therefore, it is concluded that potential damage to the tubesheet and tubes resulting from impacts by a failed thermal sleeve would not violate the integrity of the components nor create a safety concern.

4.2.3 Tube-to-Tubesheet Weld

Impacting of the tube-to-tubesheet (TTS) weld by a loose thermal sleeve may occur since the welds are partially exposed. However, complete disintegration of the weld is unlikely due to the ductility of the thermal sleeve material and the geometry of the weld. One design feature of the Model D2 steam generator is the rolled expansion of the tube over the entire depth of the tubesheet (approximately 22 inches). This feature provides added strength to the tubes in the tubesheet hole and provides an additional margin against primary to secondary leakage.

If it is assumed that some of the welds do completely disintegrate and primary to secondary leakage occurs, the amount of leakage would be low. Such leakage would be detectable by normal radiation monitoring, and the extent of the leakage could be monitored. This leakage would be expected to be within the allowable technical specification limits and would present no safety concern. Monitoring of the leakage would be possible so that if an increase is detected the plant could be shut down in an orderly manner.

4.2.4 Divider Plate

The 2 inch thick Inconel 600 divider plate is welded both to the channel head and the tubesheet to form a barrier separating the hot leg and cold leg of the steam generator. The rigidity of the plate is highest closest to the welds, and it becomes more flexible toward the middle of the plate. Impacting of a thermal sleeve could be expected to occur in the flexible region of the plate, due to the geometry of the channel head limiting access to areas closest to the welds.

The flexibility of the plate in the most likely impact region along with the flexibility of the thermal sleeve will cause the impact loadings to be sufficiently distributed so as to be of no concern to the integrity of the divider plate.

As mentioned previously, the ductility of the sleeve material would reduce the likelihood that sharp edges would be created. Therefore, any marks that result from thermal sleeve/divider plate impacts would most likely be round-bottom, rather than sharp-pointed. It is therefore unlikely that stress riser areas would be created.

The effect of impacts near the welds of the divider plate are discussed in the next section.

4.2.5 Tubesheet-to-Divider Plate and Divider Plate-to-Channel Head Welds

By the location of these welds, the amount of direct impacts they would absorb from a loose part is extremely low. In previous circumstances involving loose parts on the primary side of the steam generator at other plants, inspection of these welds showed no evidence of degradation due to impact forces.

Long-term fatigue induced by forces being transmitted to the welds by continual impacting of the divider plate and/or channel head in the region close to the welds is of no concern. As mentioned previously, with limited access to these areas due to geometry, the probability of a large number of impacts occurring close to the welds in low. Since plant operation would not continue with a loose part in the steam generator, fatigue is not a factor.

4.2.6 Channel Head

The inside of the channel head is weld clad with 308 stainless steel. This material, like Inconel 600 and 304 stainless, is ductile. Impacting of the thermal sleeve on the channel head would thus not cause any sharp dents where a point of stress concentration would form. The ductility of the 308 material makes it unlikely that enough inpacts will occur on a particular spot to cause cracking and loose cladding. Therefore, inpacting of the thermal sleeve would not be expected to adversely affect the channel head and cladding.

4.2.7 Conclusions

The potential entry of a 14 inch thermal sleeve from the pressurizer surge line into the primary side of the steam generator is not expected to cause any severe effects that would adversely effect the continued safe operation of the unit.

4.3 REACTOR INTERNALS

The reactor internals were evaluated to determine the effects of impact and wedging loads on reactor guide and support and structures due to the presence of loose thermal sleeves in the reactor coolant system.

4.3.1 Upper Internals

The 3 inch charging injection line thermal sleeves and the 10 inch safety injection/accumulator injection line thermal sleeves will be confined between the lower core plate in the reactor vessel, and the steam generator cold leg plenum. As such these thermal sleeves will have no impact consideration on reactor upper internals. The thermal sleeve located in the 14 inch hot leg pressurizer surge line does have the capability of becoming lodged in the upper internals. In a back flow or alternate leg blowdown situation if the surge line thermal sleeve became loose it could travel back through the loop and into the upper internals. The following assessment utilizes plastic analysis to determine impact loads on support columns and guide tubes in the reactor upper internals.

Support Columns

Length 80"
O.D. 5"
I.D. 3.01"

Thickness = 0.995"

$A = 12.52 \text{ in}^2$

Material : ASTM A 479 Type 304 stainless steel, cold finished.

Guide Tubes 17 x 17

Length 96"

Thickness 0.25"

Size 7.34" x 7.34"

$A = 7.09 \text{ in}^2$

Back Flow Velocity

Mass back flow 2500 lbm/sec

Density 20 lbm/ft^3

Area 4.587 ft^2

V - 27.3 ft/sec

Objects in the bottom of the reactor vessel would not be expected to reach the upper internals due to the filtering action of the fuel assemblies. The close spacing of the rods, the configuration of the grids and the flow deflectors, and the configuration of the nozzles should prevent large particles and most other particles from reaching the upper internals. Small particles which could pass through the fuel assemblies are likely to pass through the upper internals or to be forced clear during operation of the drive line. In order for a foreign object to cause interference, it would have to be preferentially oriented in a moving clearance area.

As part of the normal startup tests, control rod drop times are recorded and evaluated to confirm proper driveline performance. In the unlikely event that a foreign object would become lodged in the upper package during operation and cause a driveline to become inoperable, the existing FSAR analyses assumption of one stuck control rod assembly would not be exceeded.

The low probability of a foreign object causing a malfunction of a driveline component could be enhanced by rod stepping operations during plant operation to demonstrate continued proper driveline operations.

UPPER INTERNAL STRESS SUMMARY

	LOAD (KIP)	ALLOWABLE LOAD (KIP)	SAFETY MARGIN	DEFLECTION (INCH)
Support Column	21.6	29.1	0.34	0.238
Guide Tube	18.3	29.4	0.60	0.335

Impact loadings on reactor internals upper support columns and guide tubes have been shown to be acceptable.

4.3.2 LOWER INTERNALS

The reactor vessel and lower internals were analyzed for structural integrity with thermal sleeves from the 3 inch charging line and 10 inch SI/accumulator lines within the reactor vessel. The thermal sleeve from the 14 inch pressurizer surge line is unable to reach the reactor coolant system. The components analyzed were 1) Core Barrel, 2) Irradiation Specimen Guide, 3) BottomInstrumentation Penetration.

Impact Load Evaluation Core Barrel

Assume that the 14 inch side of a 1/4 sleeve strikes the core barrel at the inlet nozzle velocity. Since the part is thin it will deform before the core barrel. Therefore, the load applied to the core barrel is determined by the load capacity of the piece

Assuming an ultimate strength of 63.5 ksi for the piece, the maximum load applied to the core barrel is 127 kips

Assuming the core barrel responds as a cantilever beam, the impact stresses in the core barrel are calculated to be negligible.

σ_{max} = 611 psi and τ_{max} = 257 psi)

Assume that the face of a complete sleeve strikes the core barrel at the inlet nozzle velocity.

Area of piece = 4.166 in^2

Maximum load applied to core barrel is

$$P_{max} = 265 \text{ kips}$$

$$\sigma_{max} = 611 \text{ psi}$$

$$\tau_{max} = 257 \text{ psi}$$

Due to the low magnitude of the impact stresses and the short time duration of impact loads, the core barrel is unaffected by impacting loose parts.

The method used for the minimum missile energy required to perforate a target plate per WCAP 9934 results in a maximum depth of dent equal to .0234 in.

4.3.3 IRRADIATION SPECIMEN GUIDES

The irradiation specimen guides are bolted to the outside of the neutron shield panels. Each guide consists of an upper and a lower portion.

The upper portion is attached to the upper neutron panel by (six) 3/4 inch cap screws and (two) 7/8 inch diameter dowel pins. The lower portion of the irradiation specimen guide is attached to the lower neutron panel by (eight) 3/4 inch cap screws and (two) 7/8 inch dowel pins.

Since the upper portion of the specimen guide is supported by fewer bolts than was chosen for the impact load evaluation.

The assumptions pertaining to the impact load calculation are:

1. Rigid target and elastic missile

2. Loose part moves at fluid velocity

3. Maximum dynamic load factor = 2.0

The contact area is calculated assuming that the face of the thermal sleeve impacts the top of the specmen guide. The width of the top of the specimen guide is obtained from Drwg. 1094E34 Sub 6 and is 7.18 inches.

The impact load, Q, is then calculated assuming a dynamic load factor of 2.0.

$$Q \quad = 2AP = 116{,}859 \text{ lb}$$

The load required to overcome the friction due to preload and move the specimen guide is calculated to be 15,577 lbs. per bolt..

Since the impact load is much greater than the friction load, the specimen guide will move and the load will be carried by the (six) 3/4 inch bolts and dowel pins.

Since the dowel pins are installed at an interference fit and the bolts have 0.2 inches of clearance on each side, the dowel pins must fail before any significant loads may be transmitted to the bolts. Therefore, assuming that all the load is carried by the dowel pins, the shear stress in the dowel pin is calculated to be 97,221 psi.

Since the operating stresses are negligible, the fatigue usage factor is determined for impact loads only.

The shear stress is multiplied by the shear shape factor in order to include the peak stress component, for a solid circular cross section this factor is 4/3.

$$(S_s)_{max} = \frac{4}{3}\,(97{,}221) = 129{,}628 \text{ psi}$$

The stress intensity is 2 $(S_s)_{max}$ = 259, 256 psi

The alternating stress range is 1/2 (259,956) psi = 129,628 psi

Using Figure I-9.2 of Aplpendix I of Section III of the ASME Code, the allowable number of cycles for this stress range is approximately 500.

The maximum postulated number of impacts is 24, assuming that all sleeves break up into quarter sections.

Thus the cumulative usage factor for the bolts due to impact loading is

$$n = \frac{24}{500} = 0.048$$

Since n < 1.0, the postulated impacts will not cause failure of the irradiation specimen guides.

4.3.4 BOTTOM MOUNTED INSTRUMENTATION TUBES

The instrumentation tubes in the bottom head of the vessel were evaluated for impacting of thermal sleeves or thermal sleeve sections which may be loose in the system. The cases evaluated were for an impact at the tube/bottom head intersection (shear strike) and for an impact at the highest point on the instrument tube which could be struck without first striking the internals. Resulting values were compared to appropriate shear allowable and collapse loads.

The shear strike was evaluated only for the largest thermal sleeve (complete 10 inch thermal sleeve) which could impact the instruement tubes. The maximum shear stress was found to be only 1.23 KSI which gave a safety factor of 11.5 compared to the allowable of 0.6 Sm.

The loads on the instrument tubes resulting from the impacting of the complete 10 inch thermal sleeve, a one-quarter section of the 10 inch thermal sleeve, and the complete 3 inch thermal sleeve were all evaluated as exceeding the instrumentation tube collapse load, when the sleeves are assumed to strike the tubes at full downcomer velocity as they enter the lower RV plenum. This result indicates that plastic deformation of an instrumentation tube could result if the tube were struck in an unfavorable manner by the loose thermal sleeves as they enter the lower plenum. However, deformation of the tubes does not constitute a safety concern. Due to the ductility of the Ni-Cr-Fe alloy tube deformation could occur, but the tubes will not rupture and will continue to protect the thimble guide tubes. The guide will therefore not rupture and the pressure boundary will not be violated. This evaluation is supported by the experience of another plant where the presence of four 10 inch sleeves in the lower RV plenum caused no significant damage.

The loose thermal sleeves striking the instrumentation tubes in the bottom head of the vessel does not constitute a safety hazard.

In the unlikely event that the failure of a bottom mounted instrumentation tube leads to leakage, the double ended break of this BMI tube results in a leak area of 0.00024 Ft^2. Assuming a discharge coefficient of 1.0 and the Zaloudek subcooled critical flow model which overpredicts leak flow, one charging pump in the normal charging mode can provide makeup for at least 3 broken tubes.

This would be classified as a leak, not a LOCA, and RCS pressure would be maintained at 2250 psia. If both charging pumps are available, additional leaking tubes can be tolerated.

Small break LOCA analysis with minimum safeguards SI have demonstrated that full instrument line breaks in at least 5 instrument tubes may result in depressurization and automatic SI initiation. However, this small break LOCA will maintain forced or natural circulation, and the RCS will reach equilibrium conditions with no core uncovery.

4.4 REACTOR VESSEL

During plant heatup, the gap between the reactor vessel bottom head inside surface and the bottom of the secondary core support structure will decrease. Similarly, the gap in the radial support keyways will decrease during heatup. A foreign object present in this area could impose mechanical loadings on the vessel bottom head. These wedging loads were considered by determining the stiffness of a wedged cylinder (assumed intact thermal sleeve) and computing a spring rate.

The calculated spring rate and resulting loads shows that loadings from such objects would be acceptable.

Impact loads on pressure boundary components of the vessel would be low due to the low radial flow velocities in the direction of the vessel walls. These loads would be acceptable.

4.5 NUCLEAR FUEL

Foreign objects in the primary system have two potential effects on the nuclear fuel: 1) partial flow blockage of fuel assemblies due to an object becoming wedged in the fuel assembly flow paths, and 2) clad wear due to pieces becoming lodged in the assembly or between two assemblies. Flow blockage effects are discussed in Section 5 of this report.

From a fuel mechanical design viewpoint, loose pieces should not pose an operational problem when the fuel assemblies are seated properly on the core plate. The loose pieces should be stopped by the bottom nozzle or the lower core plate due to dimensional considerations. Although highly unlikely, it is possible for a very small piece to wedge between fuel assemblies and cause fretting and/or grid damage. This is highly improbable due to the fact that space between fuel assemblies is approximately 40 mils, i.e. more than three times the thickness of the thermal sleeve material. Should a fretting mechanism cause clad failure on a fuel rod it is unlikely that any radiation release would approach the technical specification limit, and as such no safety concern would exist.

Due to the relatively large fragments expected from the thermal sleeves, the transport of loose pieces into and through the fuel assemblies is not considered possible.

4.6 REACTOR COOLANT PUMP

There are no thermal sleeves located in piping connections between the Reactor Coolant Pump (RCP) and the Steam Generator. A loose thermal sleeve can enter the RCP only when a reverse flow condition occurs in which case the plant is not operating. If this occurs a thermal sleeve or portion of one will not affect the pressure boundary integrity due to the geometry, mass and impact energy of the pieces.

An intact 3 inch thermal sleeve or similar size fragments of a larger thermal sleeve can pass through the pump internals without significant deformation.

The larger thermal sleeves would not pass through the pump diffuser and impeller during a non rotating impeller condition. During RCP startup the forward flow wuuld eject any fragments.

If thermal sleeve fragments did lodge between the impeller and diffuser in such a way as to cause interference, the material is expected to be pinched or sheared between the impeller and diffuser vanes due to the very high torque of the RCP. A consequence may be an increase in shaft vibration with continued RCP operation, i.e., no locked rotor or pressure boundary violation is expected to occur. The increased vibration may be observed by the operator for corrective action.

A similar safety evaluation of larger material (1 1/16 inch thick, 304 SS) that was postulated to enter the RCP in various size fragments was previously performed, and it also concluded that there was no safety concern.

In a summary, the loose thermal sleeves are not considered a safety concern for RCP integrity and operation.

4.7 Pressurizer

The thermal sleeves in the 4 inch spray line and the 14 inch surge line connections in the pressurizer proper are attached in a different manner than the reactor coolant piping nozzle thermal sleeves. On the pressurizer thermal sleeves the upstream end of each sleeve is welded over an arc of 45 degrees. The sleeves themselves are of larger diameter than the nozzle safe ends, thus preventing sleeve movement away from the pressurizer. The flow distribution screen inside the pressurizer at the surge line connection prevents that sleeve from entering the pressurizer. Similarly, the spray header traps the sleeve on the spray line connection.

Since the pressurizer thermal sleeve design has not sigfinicantly changed and operating experience has not indicated any failure these sleeves are not considered in this safety evaluation.

Due to their method of attachment, it is also very unlikely that these sleeves would become loose within the reactor coolant system.

Based on the most probable movement of any dislodged thermal sleeves from the 10 inch SI or the 3 inch charging lines it is extremely unlikely that any piece would cause mechanical damage or become lodged in the pressurizer inlet piping or the pressurizer.

4.8 OTHER REACTOR COOLANT SYSTEM COMPONENTS

Due to the physical separation from the remainder of the reactor coolant system of such components as control rod drive mechanisms and safety, relief and block values, no adverse effect is expected to result from loose thermal sleeves in the reactor coolant system.

4.9 AUXILIARY SYSTEMS

The possibility of the potentially loose thermal sleeves affecting the operation of other systems connected to RCS was also investigated in this safety evaluation. The evaluation below considers each thermal sleeve location and the possible paths to systems or components interfacing the RCS.

4.9.1 SURGE LINE THERMAL SLEEVE

The surge line is located in loop B directly upstream of the steam generator. Assuming the sleeve came loose, it would normally travel to the steam generator. With reverse flow, the sleeve may travel to the reactor vessel, impacting along the way the three scoops for the RTD manifold. Upstream of the RTD scoops is a well (holding a temperature device) and the safety injection hot leg injection nozzle. Neither the scoops nor the well should be damaged due to the heavy grade construction of both. Likewise, the SI injection nozzle (6 inch, Sch. 160 pipe, 5.187 inch ID) should not be damaged. Entry into the injection line would be difficult if not impossible, due to the stagnant or forward flow conditions in the line.

4.9.2 Safety Injection Cold Leg Injection Line Thermal Sleeves

There are four injection lines, one in each cold leg. The normal flow directions in all loops is toward the reactor vessel. Each loop has only one interface in this direction prior to reaching the reactor vessel; the injection line from the centrifugal charging pump/boron injection tank. This line is 1 1/2 inch, Sch 160, 1.338 inch I.D. A thermal sleeve from the 10 inch cold leg injection nozzles chould not (whole or in parts) damage or enter these lines, due to the size of the sleeves and sleeve segments.

4.9.3 Normal and Alternate Charging lines Thermal Sleeves

The normal charging line enters loop A upstream of the 10 inch SI and 1 1/2 inch centrifugal charging pump/boron injection tank lines. The thermal sleeve could conceivably enter into the SI line during normal flow. However, the SI connection is above the horizontal and has stagnant flow characteristics. It is thus unlikely that a part entering the SI line would travel far. Approximately 5 feet upstream in the line is a 10 inch, normally closed check valve. The part should not migrate far enough into the line to make contact with the check valve due to the stagnant conditions in the line. If it were to remain in the line, it would be flushed into the RC pipe on SI initiation.

With reverse flow, the thermal sleeve could traverse upstream and enter into a pressurizer spray line. It would then have to travel upward against gravity in order to reach and conceivably, damage the pressure control valve. At worst, this could force the plant to a premature shutdown if the valve were to stick open: no safety significance. Should the valve stick closed, the alternate spray path would be available.

The consequences of the alternate charging thermal sleeve (or parts) migrating would be the same as the normal charging sleeve, but with no potential involvement of the pressurizer spray line.

Upstream from the safety injection lines loop dependent configurations exist. All loops have a resistance temperature detector mounted in a well and a single RTD cold leg connections. Neither of these should be affected, due to the construction of these components.

Other connections include the charging (loop A), alternate charging (loop D), pressurizer spray (loops A and B), and the excess letdown (loop C) lines.

The two charging lines, normal and alternate, are 3 inch, Sch 160 with thermal sleeve of their own. Should the SI thermal sleeve contact one of these, it may be damaged (i.e. dislodged) creating additional loose part(s) and in fact these parts have been considered in this safety evaluation. Entry into the charging lines would be difficult if not impossible due to the small I.D. (2.12 inch w/sleeve, 2.69 inch without sleeve).

The two pressurizer spray lines are 4 inch, Sch 160, 3.438 inch I.D. Damage to or entry into these lines would be difficult if not impossible, due to the size and construction of the lines.

The excess letdown line is 1 inch, Sch 160, 0.815 inch I.D. Again, entry into or damage to the line is considered improbable, due to the small line size and heavy construction of the line.

In the above discussions, it should be noted that the connections (with the exception of the RTD HL scoops) are in the upper half of the RC piping and the loose sleeve would be required to move against gravity.

It is conceivable that a failed sleeve part could enter into its own nozzle, however, flow in these branch lines is toward the reactor coolant pipe with the exception of the pressurizer surge line. In that case, the part would have to move against gravity and greatly reduced coolant flow in order to migrate far into the surge line. If it were to reach the pressurizer, the part would be trapped by a screen at the internal surge line nozzle.

4.10 MATERIALS

No unacceptable material would be introduced into the reactor systems as a result of the failure of a thermal sleeve. Minor clad damage could occur on the surfaces of carbon steel components, however, this would present no safety or operational concern due to the very slow corosion rate of the carbon steel in the reactor coolant environment.

5.0 FLOW BLOCKAGE

5.1 INTRODUCTION

In postulating the presence of loose thermal sleeves in the reactor coolant system, an evaluation was made of the effect of the sleeves or parts of the sleeves blocking flow in the core and in various locations in the reactor coolant system. The evaluation considered that all thermal sleeves come loose in the reactor coolant system, break into quarter segments, by RCS flow to the following locations:

A. The 3" sleeves and 10" sleeves in the cold leg lodge in the lower internals and block flow at the lower core plate.

B. The 14" sleeve from the pressurizer surge line blocks flow at the steam generator tube sheet. (The case of the intact 14" sleeve partially blocking flow in the hot leg was also analyzed, however, blockage at the steam generator tube sheet was determined to be more conservative).

The evaluations considered the effect of blockage on reactor coolant system, total flow, LOCA flow distributions in the core during normal operation, and the effect on LOCA and non-LOCA accident analyses.

5.1 REACTOR COOLANT SYSTEM TOTAL FLOW

For the analysis of reactor coolant system flow reduction, the loose 3" and 10" thermal sleeve segments in the reactor vessel were modeled as flat plates normal to the flow, resulting in increased pressure loss coefficients across the lower core plate.

The 14" sleeve segments in the steam generator were assumed to completely block flow to the tubes covered by the segments. This is a very conservative assumption since it is extremely likely that the segments will retain their curvature and only cause a flow restriction rather than total flow blockage.

The results of this conservative analysis showed that the total reduction in RCS flow was approximately 0.6 percent. This still results in the RCS flow being greater than thermal design flow, which is a conservatively low value of flow rate upon which the core thermal-hydraulic design is based. Thus, this flow reduction will have no effect on the thermal-hydraulic design and DNB margin in normal operation at rated power. Based on the above evaluations it was concluded that the reduction in RCS flow would not affect design margins in normal operation.

5.2 LOCAL CORE FLOW DISTRIBUTION

The effect on core flow distribution of loose thermal sleeve segments located underneath the lower core plate was also evaluated. It was determined that the segments from the sleeves remaining below the lower core plate would result in greater core blockage than the smaller segments reaching the fuel nozzles, since the smaller pieces could only reach the fuel nozzles in a lengthwise orientation. In performing this evaluation, it was assumed that the sleeve segments remain curved, and thus do not completely block flow, but do cause restrictions in the flow to the core.

The information available on thermal effects due to flow blockage indicates that there will be no significant increase in the likelihood of DNB at normal operating conditions. WCAP-7956 shows results from a blocked assembly flow recovery test and WCAP-8054 shows that a 10 percent flow reduction in the hot assembly and its 8 surrounding assemblies reduces DNBR by only 0.3 percent. Since the thermal sleeve pieces will remain curved, there will always be some flow through all of the lower core plate holes. This, along with the fact that the total core thermal design flow will remain unchanged, will insure that the DNBR will not be reduced by more than a few percent.

Thus, the effect of blockage on local core flow distribution and DNBR is judged to be insignificant.

5.3 NON-LOCA TRANSIENT ANALYSIS

Flow blockage by loose thermal sleeves in the reactor coolant system potentially affects non-LOCA transients only in that there is a slight reduction in total RCS flow, as discussed previously in Section 5.1.

An evaluation was performed on the effect of the RCS flow reduction on the non-LOCA transients. In non-LOCA transient analysis, it is conservatively assumed that accidents are initiated with the reactor coolant system operating at thermal design flow (TDF). A reduction of 0.6 percent due to the thermal sleeve flow blockage effect on RCS flow still results in a measured flow greater than TDF. This assures that all the current safety analyses remain valid.

5.4 LOCA EVALUATION

The postulated presence of loose thermal sleeve segments in the RCS was also evaluated for its effect on the 10CFR50.46 Appendix K limiting case ECCS analysis. For McGuire, the limiting case break is a double ended cold leg guillotine break with a discharge coefficient equal to 0.6. A summary of the evaluation is given below:

A. Overall system thermal performance at 100 percent power has been shown to be insignificantly changed by the presence of the large sleeve pieces. The reduction in RCS flow of 0.6 percent can be accommodated with little effect on the Appendix K LOCA analysis.

B. Considering the presence of thermal sleeve segments lodged against the bottom of the lower core plate, it is assumed that the thermal sleeves will remain curved. Thus, there will always be some flow through all of the lower core plate holes, such that no assembly will be starved of flow. It has been determined that flow redistribution above postulated sleeve location will occur in the first several inches of the fuel during normal operation, and that therefore reduced minimum DNBR is not of concern in the hot assembly. In a LOCA analysis, post-LOCA thermal-hydraulics predicted for the hot assembly directly define the calculated peak clad temperature (PCT). Core flow post-LOCA is characterized by positive (normal direction) and negative core flow periods, in that order. From the above, during positive core flow when RCP performance determines flow magnitude and direction as during normal operation, thermal-hydraulics should be equivalent to those computed in the existing LOCA analysis. When the flow reverses, any pieces impinged against the core plate should fall off into the lower plenum and thus not be in a position to affect the calculated core flow. The McGuire Plant is equipped with upper head injection (UHI) which provides ECCS water directly to core to quench the fuel rods following a large break LOCA. Performance of the UHI system will not be impaired by the presence of loose thermal sleeve material in the vessel lower plenum, so the capability for quenching and effective cooling of fuel rods during blowdown remains.

C. It has been assumed, in most areas of this safety evaluation that the thermal sleeves break up into quarter segments, however the presence of smaller pieces of the sleeve being lodged within the fuel providing additional blockage in the hot assembly during core reflood following a LOCA was also addressed.

In the limiting case LOCA analysis for McGuire, the maximum calculated peak clad temperature (PCT) occurs immediately after the bottom of the core is recovered. At this time, a high flooding rate is in effect, Appendix K to 10CFR50 requires a fuel blockage flow penalty to be considered during reflood at low flooding rates (below 1"/sec). In McGuire, this low flooding rate occurs later in the transient after the PCT is reached. Thus, the postulation that added blockage from the thermal sleeve pieces will cause added fuel blockage will not affect the calculated McGuire peak clad temperature in a LOCA transient, since the PCT is reached prior to consideration of blockage effects.

D. The possibility that the hot leg might contain loose parts due to the dislodging of the thermal sleeve in the pressurizer surge line nozzle was evaluated. Due to the plethora of guide tubes, support columns, etc. in the upper plenum it is not possible that any piece could orient itself in such a way as to significantly block flow exiting any particular fuel assembly which is located beneath either a guide or support column. Since only low power peripheral assemblies are not located below a support column or guide tube in the McGuire plant, any pieces in the hot leg are of no concern from the standpoint of the ECCS analysis.

E. Another area addressed in the LOCA evaluation is the possibility that a pressurizer sleeve propelled by post-LOCA blowdown forces might impact and damage a particular guide tube or support column in the upper internals. In the Westinghouse UHI Evaluation Model the conservative assumption is routinely made that no flow from the upper head is delivered directly into the hot assembly containing the hot fuel rod. Rather, upper head water which enters guide tubes is delivered to the upper plenum, and water which flows through the support columns is assumed to enter the core exclusive of the hot assembly. In addition, no credit is taken in the ECCS analysis for operation of the control rods to shut down the plant. Thus, failure of a guide tube/support column will have no significant impact on the McGuire Plant ECCS performance analysis although no damage to control rods is expected.

F. Finally, the blockage of steam generator tubes in the loop containing the pressurizer surge line was also considered. In this case, it is assumed that segments of the pressurizer surge line thermal sleeve are held against the steam generator tube sheet by reactor coolant pump flow prior to a LOCA. In this situation, during the initial part of the LOCA transient when the RCS is still in forward flow due to the influence of the RCPs, the core thermal-hydraulics should be equivalent to the existing LOCA analysis. When flow reverses, and the steam generator channel head voids, the thermal sleeve pieces held against the tube sheet will fall off into the channel head and not be in a position to affect flows in the reactor coolant system. Thus, the effect of the postulated sleeve segments at the steam generator tubesheet will not significantly affect the 10CFR50.46 Appendix K ECCS analysis.

Attachment 2

THERMAL SLEEVES - INSPECTION RESULTS

During recent shutdown for steam generator inspection, Duke Power received information regarding the thermal sleeves found in another Westinghouse plant. As a result, an inspection of the four 10 inch thermal sleeves for the cold leg accumulator injection lines was performed.

The inspection was performed by radiographing the 10" lines in the nozzle area. Starting July 1, these inspections proceeded on loops A, C, and D with all thermal sleeves found in place. On July 5, the radiograph of loop B indicated the thermal sleeve was missing. THis was confirmed by a visual inspection with a small TV camera going through the upstream check valve on the 10" line.

This discovery prompted an inspection of other connections to the RCS having similar thermal sleeves. The three inch lines were radiographed on July 6 and all thermal sleeves were found in place with their welds intact. The 14" line was examined initially to determine the presence of the thermal sleeve. Additional radiography on the 10" and the 14" lines on July 7 and 8 indicated that the welds on the these thermal sleeves showed no indication of degradation. The 14" line contained reactor coolant water at this time which degraded the sharpness of the image. However, after several exposures, no obvious defects were noted.

Additional visual inspections of the loop B 10" line indicated the welds located at the top of the sleeve had failed at the interface to the nozzle wall with possibly only a small portion of one weld remaining on the nozzle wall. The nozzle wall showed no indication that the sleeve had broken apart prior to being released.

On July 10, with residual heat removal flow through Loops A and B, an indication of minor metal impacts was indicated by the loose parts monitor. Analysis indicated this was a large metal object probably rolling against the lower internals structure. Subsequent reactor coolant pump runs for filling and venting of system substantiated that a large metal object was located in lower reactor vessel internals. Running the reactor coolant pump in Loop B produced the most significant movement while with more than two reactor coolant pumps running, there was no movement. At no time were these impacts above the alarm setpoint of .5 ft/lbs.

As a result, it is concluded with reasonable assurance that the missing 10" thermal sleeve is located in lower reactor internals and the small impacts during low flow conditions are of a minor nature. No movement is indicated when full flow is present indicating the sleeve remains fixed in place.

INSPECTION PROGRAM SUMMARY

July 1	-	Radiographic examination of 10" lines began.
July 1-4	-	RT results indicated thermal sleeves in place on Loops A, C, and D.
July 5	-	RT results on Loop B 10" line indicated thermal sleeve missing. Confirmed by visual inspection early on July 6.
July 6	-	Radiographic examination on 3" lines on Loop A and D showed sleeves in place and welds intact.
July 7	-	Additional visual inspection of Loop B 10" line showed no sleeve pieces remaining in nozzle.
July 7	-	Radiographic examination of 14" line on Loop B showed sleeve in place and welds intact.
July 8	-	Additional RT on remaining sleeves in 10" lines showed welds intact.
July 10	-	During operation of decay heat removal system minor impacts in lower reactor vessel internals.
July 11	-	During reactor coolant pump runs minor impacts with one pump running, no movement with all 4 pumps running.

ATTACHMENT 3

Proposed Plans for Unit Operations with Detached Thermal Sleeve

Introduction

Duke Power Company has concluded that operation of McGuire Unit 1 with detached thermal sleeves does not present a significant hazard to the public health and safety. Added assurance to support this conclusion is provided by the capability to detect any impacts caused by loose parts so that appropriate actions can be taken to prevent damage to RCS components which may conceivably be caused by repeated impacts. The loose parts monitoring system described below provides this capbility. In addition, increased surveillance of certain plant parameters will provide additional assurance that no damage to components in the reactor will occur without detection. Therefore, Duke Power Company proposes to return McGuire 1 to power operation.

LOOSE PARTS MONITORING SYSTEM

The Loose Parts Monitoring System is designed to detect and record signals resulting from impacts occuring within the Reactor Coolant System. Eight transducers are located in the areas where loose metal objects are most likely to become entrapped.

These are:

- two on the reactor vesel lower head, diametrically opposed.
- two on the reactor vessel upper head, diametrically opposed.
- One on the lower head of each of the steam generators.

Experience has shown that the exact location of these accelerometers is not critical since the acoustic wave resulting from an impact propagates through-out the entire head.

The Loose Parts Monitoring System will detect any object striking the Primary System with an impact energy of .5ft.-lbs. (equivalent to the impact of a .5 inch stainless steel cube with a velocity of 30.2 ft/sec). The impact is measured in terms of "acceleration" by the use of transducers that convert acceleration to an electrical signal. A transducer contains a piezoelectric material fabricated to provide for changes in compression of the material in response to accelerations. A small electric charge proportional to the acceleration is generated by piezoelectric action. This change is converted to a voltage signal by a line driver preamplifier. The line driver voltage is then treated as a normal instrument signal requiring normal shielding and cable considerations. An impact calibration is performed during system installation. This calibration establishes the relationship between the output signal strength and impact energy. It also determines the frequency of the damped ringing signal that is characteristic of all large steel structures when struck. To perform the impact calibration, the loose parts monitoring system is activated and the reactor vessel and steam generators are then struck with a known impact energy. The locations for the impacts are selected at the time of impact calibration. Selection is based primarily upon accessibility.

The data acquisition panel inclused a visual-audible alarm system to annunciate if any channel exceeds its alarm setting. Separate indication lamps are provided for each channel. An on-line F.M. tape recorder capable of recording any four channels simultaneously with an automatic turn on to record any channel (up to four simultaneously) that exceeds its preset alarm level. Additional capabilities of the system provided input for spectrum analyzer and strip chart recorder to produce permanent on-line P.S.D. plots to aid in forecasting and preventive maintenance.

Since the detached thermal sleeve was discovered, the LPM system calibration has been verified by the manufacturer. This verification assured all eight channels were fully operational.

OPERATING PROCEDURE REQUIREMENTS

If during plant operations, the operators receive an alarm from the LPM, the possibility exists that an additional thermal sleeve has broken free and has been swept into the steam generator (in the case of the 14" pressurizer surge line) or the reactor vessel. A procedure has been revised to include guidance that in case of an alarm in the steam generator, the plant will be brought to cold shutdown as quickly as practical. In addition the loose part will not be allowed to potentially enter the reactor vessel/upper internals which may result in damage and complicate the retrieval of the loose part. To assure this the reactor coolant pump in the loop with the alarm will be the last pump to be stopped.

In the case of an alarm in the lower reactor vessel, the plant will be shutdown in a controlled manner if the alarm indicates a loose part. Evaluations will be made to determine if indicated movement is different than previously evaluated.

In addition, monitoring of certain parameters will be increased to insure detection of any change in operating parameters. These include increased surveillance frequencies for:

- control rod movement test
- incore detector map
- iodine concentration in reactor coolant.

OPERATOR TRAINING

The revised procedure has been covered with all operating personnel. The shift Technical Advisors have been briefed on the potential for loose part indications and will provide technical assistance to plant operators. Other plant personnel (reactor engineering, chemistry, health physics) have been briefed on findings and necessity for increased survelliance in areas noted.

SCHEDULE FOR PLANT OPERATION

Duke Power Company is in the process of preparing the unit for start up following the current outage associated with the steam generator inspection. We intend to operate the unit at power levels below 75% power until appropriate modifications are made to the steam generators. This is an additional margin of safety relative to the concern of the thermal sleeves.

It is Duke Power Company's intent to remove all of the installed 3", 10", and the 14" thermal sleeves and retrieve the loose 10" thermal sleeve by the first refueling outage. If an outage of sufficient duration occurs earlier, the sleeves will be removed at that time.