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SAFETY EVALUATION FOR

QUAD CITIES UNIT 1

CORE SPRAY LINE REPAIR

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TABLE OF CONTENTS

1.0 INTRODUCTION

The semi-circular core spray line (CSL) is a 304 stainless steel piping run internal to the reactor vessel. Its purpose is to carry the core spray system flow from the core spray nozzle thermal sleeve to two of the core spray sparger junctions that are part of the shroud assembly. The 6-inch CSL laterals are welded to a 8.625 inch outside diameter T-box. Horizontal sections of the CSL on each side of the T-box are supported from the vessel wall by a CSL bracket which is welded to the vessel 20 inches from the nozzle, and a CSL clamp located at 65°, 125°, 245° and 305°. During the 1994 refueling outage a circumferential IGSCC defect was found at the outside surface of the right hand CSL 6-inch lateral where it is welded to the CSL T-box of the 185° azimuth (N5B) nozzle. The crack was visible for approximately 180 degrees, and was in the weld heat affected zone material about 1/4 inch from the weld. A repair (by modification) was designed that addresses the identified and potential further cracking adjacent to the CSL-to-T-box welds at the N5B nozzle.

2.0 SUMMARY

The core spray line repair clamp is designed to assure that the existing core spray line will perform its function of directing ECCS flow to the core spray spargers. The core spray line repair clamp has been designed for the remainder of the 40 year plant life. The core spray line repair clamp was designed using materials and stress limits which will assure adequate margins for all design basis conditions. The hardware is designed using ASME Boiler and Pressure Vessel Code, Section III, Subsection NG stress limits. The materials used to fabricate the repair clamp hardware were selected to provide the required strength and compatibility with the reactor environment.

Leakage through an assumed 360° crack at the identified crack location with the CSL modification installed was considered; it is concluded that the leakage will conservatively increase the peak clad temperature (PCT) by less than 40°F. However, the PCT increase is small, and when added to the existing PCT is significantly less than the regulatory limit of 2200°F per 10CFR50.46. Finally, there will be no change in the scenarios for accidents or anticipated transients.

3.0 RESPONSE TO 10CFR 50.59 QUESTIONS

3.1 Documents Implementing the Proposed Change.

a. FDI No. 325-63060

b. Drawing. 107C5499G001

c. Parts List PL107C5499G001

d. Design Specification 25A5561

3.2 Description and Purpose of the Proposed Change.

The function of the modification is to assure the structural integrity of the core spray line even if the reported defect were to grow to the full circumference of the pipe. The proposed change adds two clamps which straddle the core spray line T-box . The repair is shown conceptually in figure 3-1 The saddle clamps which are bolted through the CSL piping arms are held together with four preloaded tension members (rods). The plate which bears against the T-box cover plate prevents the T-box and thermal sleeve from moving inward, even if both lateral connections with the T-box were to be completely separated from the T-box.

3.3 The proposed change is permanent.

The proposed change is designed for the remainder of the 40 year plant life using the ASME Boiler and Pressure Vessel Code, Section III, Subsection NG (1989 Edition) as a guide for design and analysis. The repair clamp hardware is classified as safety-related, and is designed to currently accepted standards. Therefore, it can withstand the same design bases loads as the current core spray line under normal and abnormal operating conditions. The installation of this hardware will not affect (degrade) the other RPV internals.

3.4 List of SAR sections which describe the affected systems, structures or components (SSCs) or activities. Also, list the SAR accident analysis sections which discuss the affected SCCs or their operation. List any other controlling documents such as SERs, previous modifications or Safety Evaluations, etc.

SSCs & Accident Analyses	SAR Sections
Core Spray System	6.3, 3.9.5
Core Spray Safe End Replacement	*none*
Loss of Coolant Accident (LOCA)	15.6.5
Recirculation Pump Seizure	15.3.3

3.5 Description of how the change will affects plant operation when the changed SSCs function as intended (i.e. , focus on system operation/interactions in the absence of equipment failures). Consider all applicable operating modes. Include a discussion of any changed interactions with other SCCs.

If the current crack grows to a sufficient size, the remaining CSL pipe material will no longer be capable of holding the lateral to the T-box during a core spray system injection. The design of the modification assumes that the present cracks will eventually grow to 360°. The installation of the core spray line repair will limit the separation between the lateral pipe and the T-box to 0.042 inch. The modification will also prevent the T-box and thermal sleeve from displacing inward even if both lateral arm attachments were to fail. This maximum separation (.042 inch) results from the temporary cooling of the CSL relative to the CSL repair preloaded tension rods. It is conservatively assumed that the clamp cross bolts are still at 500°F when the CSL laterals are cooled to 195°F, average temperature, by the injected water from the torus (305°F temperature difference). After a few minutes of post LOCA core spray system injection, the CSL lateral pipe and clamp cross bolt temperatures will equilibrated, and the CSL modification hardware will again close the opening to a typical tight crack width (assumed 0.005 inch). The slip fit thermal sleeve may also experience a temperature difference relative to the thicker safe end during the initial core spray injection which will increase its clearance with the safe end.

There are several existing leakage locations in the CSL as well as the reported crack. Calculated leak flows from these, considering the initial LOCA temperature differentials, are shown in Table 1 below. This leakage would go into the reactor, outside the shroud, and would therefore, not contribute to cooling and reflooding of the core in a recirculation line LOCA accident. The minimum system capability is 4,700 gpm with the reactor at 90 psig based on the core spray system acceptance criteria specified in the station surveillance procedures. During pre-operational testing, the demonstrated minimum system flow capability was reported as 5,800 gpm. Previous LOCA analyses assumed 4,500 gpm core spray delivery inside the shroud when the reactor is a 90 psig relative to the drywell, therefore, the margin for this leakage is 200 gpm. The total leakage with the crack at maximum separation (277 gpm) exceeds the documented 200 gpm excess system capability, thus impact to LOCA analyses must be evaluated. Once the temperature differences subside, the total leakage will decrease to 61 gpm and will be within the system excess capacity, as shown in Table 2.

Table 1 - Leakage With Initial Temperature Differentials

Location of Leakage	Calculated Leakage (gpm @ 47 psid)	Calculated Leakage (% of 4700 gpm System Capability, Reactor at 90 psig)
Slip Fit Thermal Sleeve-to-Nozzle Safe End (0.030 max. diameteral gap with an initial 305°F safe end to thermal sleeve temperature difference)	124	2.6
0.25 dia. Vent Hole In CSL T-box	8	0.2
0.094 dia. Purge Hole In Nozzle Thermal Sleeve	1	0.0
Four Added Modifications Holes	16	0.3
360° Crack @ Maximum Separation	128	2.7
Total With Crack @ Maximum Separation	277	5.9

Table 2 - Leakage, Temperature Differentials Equilibrated

Location of Leakage	Calculated Leakage (gpm @ 47 psid)	Calculated Leakage (% of 4700 gpm System Capability, Reactor at 90 psig)
Slip Fit Thermal Sleeve-to-Nozzle Safe End (0.005 max. diameteral gap)	21	0.4
360° Crack @ 0.005 inch Separation	15	0.3
Vent, Purge and Modification Holes	25	0.5
Total With Crack @ 0.005 inch Separation	61	1.3

Leakage from the CSL has an impact on the ECCS performance evaluation. For a bounding leakage of 400 gpm (resulting in a CS flow of 4100 gpm injected inside the shroud compared with the rated CS flow of 4500 gpm for one loop at vessel pressure of 90 psig), the impact has been assessed to increase the peak cladding temperature (PCT) by less than 40°F compared with the current licensing basis (References 1 and 2). This assessment is based on the SAFER/GESTR analysis performed to support the safety evaluation of the reactor internals configuration for the 1994 QC 1 restart (Reference 3). The current loss-of-coolant-accident (LOCA) analysis basis yields a bounding licensing PCT of 1725°F for the design basis LOCA event (through the end of Cycle 15). The 10CFR50.46 regulatory limit PCT is 2200°F. The maximum potential impact of the CSL leakage on the design basis LOCA PCT is small compared with the margin to the regulatory limit, and applicable ECCS regulatory requirements are met. The sequence of events remains essentially unchanged for the LOCA events with the CSL leakage, and long-term cooling is not affected.

The additional flow restriction resulting from the modification clamp bolts penetrating the 6 inch CSL was calculated to be a 0.7 psi additional pressure drop. This pressure drop is small relative to the 47 psi pressure drop originally specified for the flow path from the nozzle safe end to the shroud dome area. The impact of this drop on system flow capability would be insignificant, and as discussed above a bounding evaluation using a system delivery of only 4100 gpm meets regulatory requirements.

3.6 Description of how the change will affect Equipment Failures.

During normal power operation instrumentation continuously monitors the differential pressure between a pressure tap above the core support plate and the core spray sparger (via the CSL), and an alarm will activate if the CSL were to break. The proposed CSL repair will keep the CSL lateral and T-box tightly together during normal operation so that the CSL break instrumentation function will not be impaired.

All of the events in Quad Cities Unit 1 UFSAR, December 1993, were examined to determine if the probability of occurrence of any of these events is increased by the installation of the repair clamps at Quad Cities 1. Potential failure modes associated with the use of the repair clamp have been investigated, and it has been concluded that the probability of any UFSAR event has not been increased.

The repair clamp has been designed, analyzed, and constructed to meet or exceed the original quality standards as prescribed in the UFSAR. A stress analysis has been performed using the ASME Boiler and Pressure Vessel Code Section III Subsection NG as a guide and this analysis confirms the structural acceptability of the design. The repair design minimizes the potential for intergranular stress corrosion cracking by the elimination of all welding (except for tack welds).

The repair has been evaluated for leakage for a design bases LOCA. See 3.5 above.

Consequently, it is concluded that the probability of occurrence of any UFSAR event with the repair clamp installed will not be increased when compared to the currently installed welded type 304 stainless steel with its potential for intergranular stress corrosion cracking.

The probability of loose parts from the CSL repair hardware is low. The hardware is designed using ASME Boiler and Pressure Vessel Code, Section III, Subsection NG stress limits. The threaded fasteners are tack welded twice to prevent their loosening. Flow velocity over the outside of the CSL pipe and the CSL modification hardware is not sufficiently high to expect any flow induced vibration of the repaired CSL. The vortex shedding frequency due to core spray system flow over the CSL modification clamp bolts was found to be less than one tenth of the lowest natural frequency of the clamp bolt, thus this feature, added as part of the modification, will not result in a new vibration concern.

The safety concerns associated with loose parts in the reactor are the potential for fuel bundle flow blockage and subsequent fuel damage, the potential for interference with control rod operation, and the potential for corrosion or adverse chemical reactions with other reactor materials.

All of the CSL modification parts are made from materials which are approved for in-reactor use. There is no potential for corrosion or adverse chemical reaction with other reactor materials.

The probability of the repair clamp failing in such a way that it is drawn into the recirculation suction line and then into the recirculation pump has not increased compared to other RPV internals in the same general location. Therefore, this replacement does not increase the probability of the occurrence of the recirculation pump seizure event abnormal operational transient.

Installation of the CSL modification involves Electrical Discharge Machining (EDM) which generates swarf consisting of very fine particles comprised of carbon, nickel, iron, chromium, etc. (the elements contained in the EDM electrode and the 304 stainless steel pipe material). These particles are very small (approximately 30 - 50 microns). Much of the swarf generated is collected by the EDM electrode flushing system. However, when the EDM electrode breaks through the pipe wall, the flushing system cannot collect the swarf, thus some of this swarf will remain in the reactor.

The sand-like particles from the EDM process are too fine to be caught at one of the fuel spacers. Most likely, these particles will be carried by the cooling flow up through the length of fuel bundles and then be discharged from the reactor core through the top of the upper tie plate. They will eventually be removed from the reactor coolant by the reactor water cleanup (RWCU) system. Therefore, there is no potential for fuel fretting due to the EDM process.

The potential for the particles generated by the repair processes causing CRD seal wear was also evaluated. Because the particles generated are so small, they will most likely be carried by the cooling flow up through the length of fuel bundles and then be discharged from the reactor core through the top of the upper tie plate or by the core bypass flow through the core region and then be discharged through the top guide. They will eventually be removed from the reactor coolant by the reactor water cleanup (RWCU) system. The upward flow direction makes it highly unlikely that these particles will be deposited on the top of the core plate so that they can migrate to the bottom of the control rod guide tubes where they could be sucked into the CRD. Therefore, it is very unlikely that these particles will have any significant effect on CRD seal wear or adverse effects on CRD operation.

In addition to the CRD seals, the potential for the particles generated by the repair processes adversely affecting the reactor recirculation pump seal performance or life was evaluated.

Ideally, the Reactor Recirculation (RR) pump seals should be operated in a clean, and air free environment. This objective is achieved by venting the seals after maintenance and purging the seals during operation. Seal purge injects 2 gpm of clean CRD water into the seal cartridge to keep solids such as the reactor corrosion products, or in this case the EDM by-products from reaching the critical components of the seal.

3.7 Identification of each accident or anticipated transient described in the SAR where the change alters the initial conditions used in the SAR analysis, the changed SSC is explicitly or implicitly assumed to function during or after the accident, or operation or failure of the changed SSC could lead to the accident.

None

3.8 List of Technical Specification (Safety Limit, Limiting Safety System Setting or Limiting Condition for Operation) where the requirement , associated action items, associated surveillance, or bases may be affected.

The 200 gpm leakage allowance, which is currently a basis for the core spray system surveillance test acceptance limit, is no longer appropriate given the leakage documented in this report. However, it is not necessary to change the current system surveillance acceptance limits, because the impact to LOCA analysis evaluation (less than 40°F increased PCT) is conservatively based on a 400 gpm leakage assumption taken from the original specified 4500 gpm system delivery inside the shroud. It is recommended that the current surveillance test acceptance limit be retained as is.

3.9 Will the change involve a Technical Specification Revision?

No

3.10 Response to increased probability or increased consequences questions for each accident listed in Section 3.5.

The probability of occurrence of an accident previously evaluated in the UFSAR will not be increased.

The consequences of an accident previously evaluated in the FSAR will not be increased.

All of the events in Quad Cities 1 UFSAR were examined to determine if the consequences of any of these events is increased by the installation of the repair clamps. Consequences (i.e., radiological doses) associated with the design basis accidents are evaluated in the UFSAR. The existing core spray line and the repair clamps do not function to mitigate the consequences of any UFSAR event except the design basis LOCA event. No UFSAR dose calculation will be impacted by this change. For the design basis LOCA event discussed in the UFSAR, the core spray sparger and core spray line provide the flow path inside the RPV for the ECCS flow to the core spray spargers. Maintaining this flow path is required to ensure that core reflooding capability is maintained following the design basis LOCA. A conservative assessment of the leakage through the crack in the core spray line was performed. It was concluded the leakage will result in an increase in the peak clad temperature (PCT)of less than 40°F. However, the PCT increase is small, and when added to the existing PCT is less than the regulatory limit of 2200°F per 10CFR50.46.

The probability of occurrence of the accidents considered in the safety analysis report has not changed. The main accidents considered are: rod drop accident, fuel handling accident, steam line break accident outside the containment and the design basis loss of coolant accident. Except for the rod drop accident, the other accidents produce postulated conditions that are outside the reactor vessel and do not impact the passive behavior of reactor internal components. The postulation in the rod drop accident included a control rod being stuck inside the core. The core spray components are outside the core and do not interfere with the assumptions used in the rod drop accident. Thus the probability of occurrence of a rod drop accident is also unaffected.

The consequence of the rod drop accident is not changed, because its consequence is not affected by the capability of the ECCS.

For the fuel handling accident the consequence is unaffected, because this accident is postulated when the plant is in a safe shutdown condition and the vessel is filled with water.

The consequence of a steam line break accident previously evaluated in the UFSAR is unaffected due to the repaired core spray piping. This accident, on a worst case basis, results in negligible doses to the public.

The consequence of a design basis recirculation suction break is unaffected, because the doses given in the UFSAR chapter 15 are based on Regulatory Guides, bypass leakage, standby gas treatment system efficiencies, etc., and not dependent upon core cooling capabilities.

An evaluation was performed (in section 3.5) to determine the PCT impact for the Core Spray Line repair. Utilizing conservative coolant loss assumptions, it was determined that sufficient PCT margin exists between the resultant PCT and the 2200°F limit in 10CFR50.46.

Repaired Core Spray Line components are all passive components. Malfunction of passive components means potential structural failure or postulated passive failure. The crack indication in the Core Spray Line was evaluated in Reference 3. It was concluded that the crack indication will arrest; therefore, any potential degradation is limited. Thus the repair clamp will reduce the propensity for the growth of the existing crack indication in the core spray line and assure the structural and geometric integrity of the assembly. Therefore, it is concluded that the potential for malfunction of these reactor internal components would be minimal.

The possibility of an accident which is different than any already evaluated in the UFSAR will not be created.

As discussed above, the repair clamp installation will not introduce any new failure mode, which might cause an accident. Also, no previously evaluated accident sequence will be affected.

The probability of a malfunction of a safety-related structure, system, or component previously evaluated in the UFSAR will not be increased.

As discussed above, the repair clamp installation will not introduce any new failure mode, which might increase the probability of a malfunction of a safety-related structure, system, or component.

The consequences of a malfunction of a safety-related structure, system, or component previously evaluated in the UFSAR will not be increased.

The repair clamps will not affect any fission product barrier, will not increase any radiation source term, or prevent any component from performing its safety-related function (such as a containment isolation). Therefore, there will be no method for the installation of this hardware to impact the consequences of any malfunction or accident.

The consequences of the failure of the repair clamps will not increase when compared to the consequences of the failure of the existing core spray line.

3.11 The possibility of a malfunction of a safety-related structure, system, or component different than any already evaluated in the UFSAR will not be created.

The repaired core spray components with crack indication impact the results of some of the loss of coolant accidents. These components are passive items inside the reactor vessel. All accidents considered in the UFSAR are those which may result in radiological doses to the public, such as LOCA, positive reactivity insertion, and handling of radioactive materials inside or outside the containment. However, any postulated accident caused by the repaired passive reactor internal components do not result in radiological doses to the public at the site boundary worse than the accidents already covered in the UFSAR. The potential opening of a crack in the core spray line is limited on account of the actual repair clamp geometry, and is also bounded by ECCS analysis results. Because of the repair clamp on the core spray line, the function of the core spray is restored. Consequently, the geometrical integrity of the core spray line and T-box area is stabilized.

3.12 No revisions of Technical Specifications are required for this modification, thus the detail questions related to Technical Specification are not answered.

3.13 The margin of safety as defined in the basis for any Technical Specification will not be reduced.

There is no setpoint, setpoint basis, or limiting condition of operation (LCO) that is dependent on the leak tightness of the reactor internal components such as the core spray line. Because the utility wants to be aware of any deterioration of the cracked components, there is no reason to relax any setpoint, change an LCO basis, change an action or surveillance requirement which may indicate a change in the actual performance of the degraded components.

There are core spray system surveillance requirements in the Technical Specification. During actual surveillance testing of these low pressure systems, there is no coolant injection into the reactor vessel. This means that the degraded reactor internal components are not involved in the surveillance testing of the ECCS, and therefore, they do not affect the Technical Specification requirements.

4.0 CONCLUSION

The use of the CSL repair clamp does not change the safety classification of the core spray line. Nor does its use have an affect on any Technical Specification or the UFSAR. The probability and consequences of design basis accidents are not changed by the installation of the clamp at Quad Cities 1.

Quad Cities 1 can be operated with the given repaired reactor internal components as addressed above. The potential consequences have been evaluated in detail. The known limiting consequence is the increase of PCT because of the coolant loss outside the core shroud from all sources including the existing crack indication, if a postulated recirculation suction pipe break were to occur. This increase in PCT is small and is justifiable with the current design basis of the core spray T-box repair.

5.0 REFERENCES

1. T. C. Hoang, et al., *Quad Cities Nuclear Power Station Units 1 & 2 SAFER/GESTR - LOCA Loss-of-Coolant Accident Analysis,* General Electric Company, July 1989 (NEDC-31345P Revision 2).
2. Letter, R. W. Tsai (ComEd) to R. E. Kingston (GE), *Summary of Quad-Cities LOCA Analysis Design Basis,* December 4, 1995 (NFS: BSA: 95-085, CHRON 216490).
3. D. T. Shen, *Safety Evaluation of reactor Internals Configuration for the 1994 Quad Cities 1 Restart,* General Electric Company, June 1994 (GE-NE A0005873-19A).

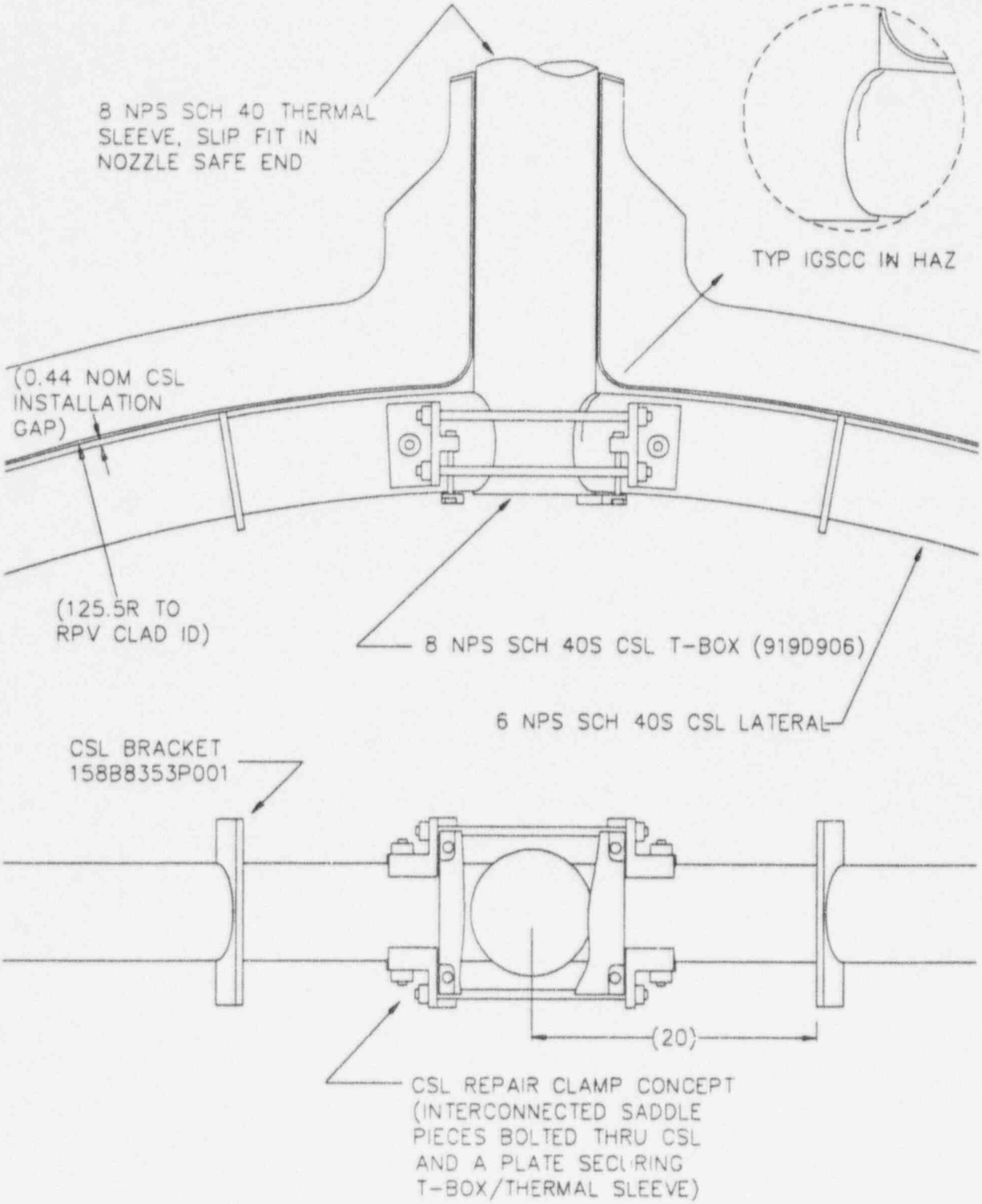

Figure 3-1 CSL Repair Concept