Florida Power
A **Progress Energy** Company

Crystal River Nuclear Plant
Docket No. 50-302
Operating License No. DPR-72

Ref: 10 CFR 50.73

November 28, 2001
3F1101-03

Document Control Desk
U. S. Nuclear Regulatory Commission
Washington, DC 20555-0001

Subject: LICENSEE EVENT REPORT (LER) 50-302/01-004-00

Dear Sir:

Please find attached Licensee Event Report (LER) 50-302/01-004-00. The LER discusses discovery of minor reactor coolant system pressure boundary leakage due to primary water stress corrosion cracks found in Control Rod Drive Mechanism nozzle #32. This report is being submitted pursuant to 10CFR50.73(a)(2)(i)(B) and 10CFR50.73(a)(2)(ii)(A).

If you have any questions regarding this submittal, please contact Mr. Sid Powell, Supervisor, Licensing and Regulatory Programs at (352) 563-4883.

Sincerely,

James H. Terry
Engineering Manager

JHT/lvc

Enclosure

xc: Regional Administrator, Region II
Senior Resident Inspector
NRR Project Manager

IE22

NRC FORM 366 (7-2001) U.S. NUCLEAR REGULATORY COMMISSION	APPROVED BY OMB NO. 3150-0104 EXPIRES 7-31-2004
LICENSEE EVENT REPORT (LER) (See reverse for required number of digits/characters for each block)	Estimated burden per response to comply with this mandatory information collection request: 50 hours. Reported lessons learned are incorporated into the licensing process and fed back to industry. Send comments regarding burden estimate to the Records Management Branch (T-6 E6), U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, or by internet e-mail to bjs1@nrc.gov, and to the Desk Officer, Office of Information and Regulatory Affairs, NEOB-10202 (3150-0104), Office of Management and Budget, Washington, DC 20503. If a means used to impose information collection does not display a currently valid OMB control number, the NRC may not conduct or sponsor, and a person is not required to respond to, the information collection.

1. FACILITY NAME	2. DOCKET NUMBER	3. PAGE
CRYSTAL RIVER UNIT 3	05000 302	1 OF 9

4. TITLE

Reactor Pressure Vessel Head Leakage Due to Control Rod Drive Mechanism (CRDM) Nozzle Degradation

5. EVENT DATE			6. LER NUMBER			7. REPORT DATE			8. OTHER FACILITIES INVOLVED	
MO	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REV NO	MO	DAY	YEAR	FACILITY NAME	DOCKET NUMBER
10	01	01	01	– 004 -	00	11	28	01	FACILITY NAME	DOCKET NUMBER

9. OPERATING MODE	5
10. POWER LEVEL	000

11. THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check all that apply)

	20.2201(b)		20.2203(a)(3)(ii)		50.73(a)(2)(ii)(B)		50.73(a)(2)(ix)(A)
	20.2201(d)		20.2203(a)(4)		50.73(a)(2)(iii)		50.73(a)(2)(x)
	20.2203(a)(1)		50.36l(1)(i)(A)		50.73(a)(2)(iv)(A)		73.71(a)(4)
	20.2203(a)(2)(i)		50.36l(1)(ii)(A)		50.73(a)(2)(v)(A)		73.71(a)(5)
	20.2203(a)(2)(ii)		50.36l(2)		50.73(a)(2)(v)(B)		OTHER Specify in Abstract below or in NRC Form 366A
	20.2203(a)(2)(iii)		50.46(a)(3)(ii)		50.73(a)(2)(v)(C)		
	20.2203(a)(2)(iv)		50.73(a)(2)(i)(A)		50.73(a)(2)(v)(D)		
	20.2203(a)(2)(v)	X	50.73(a)(2)(i)(B)		50.73(a)(2)(vii)		
	20.2203(a)(2)(vi)		50.73(a)(2)(i)(C)		50.73(a)(2)(viii)(A)		
	20.2203(a)(3)(i)	X	50.73(a)(2)(ii)(A)		50.73(a)(2)(viii)(B)		

12. LICENSEE CONTACT FOR THIS LER

NAME	TELEPHONE NUMBER (Include Area Code)
Loretta V. Cecilia, Project Engineer	(352) 563-4546

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

CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX	CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX
B	RCS	NZL	B020	Y					

14. SUPPLEMENTAL REPORT EXPECTED				15. EXPECTED SUBMISSION DATE	MONTH	DAY	YEAR
	YES (If yes, complete EXPECTED SUBMISSION DATE).	X	NO				

16. ABSTRACT (Limit to 1400 spaces, i.e., approximately 15 single-spaced typewritten lines)

At 1300, on October 1, 2001, Florida Power Corporation's (FPC's) Crystal River Unit 3 (CR-3) was in MODE 5 (COLD SHUTDOWN) at 0 percent RATED THERMAL POWER. While performing a visual inspection of the reactor vessel head (RVH), FPC personnel identified one potential leaking CRDM nozzle (nozzle #32). The RVH inspection was performed to satisfy a commitment made by FPC in response to NRC Bulletin 2001-01, "Circumferential Cracking of Reactor Pressure Vessel Head Penetration Nozzles." At approximately 1748, on October 8, 2001, CR-3 was in MODE 6 (REFUELING) at 0 percent RATED THERMAL POWER. Ultrasonic testing (UT) examination of RVH nozzle #32 identified the leakage path as two (2) axially oriented cracks that were through-wall. An 8-hour notification was made to the NRC in accordance with 10CFR50.72(b)(3)(ii)(A) due to confirmation of Reactor Coolant Pressure Boundary leakage while at power. The cracks were caused by primary water stress corrosion cracking. Eight (8) additional CRDM nozzles were examined using UT. No evidence of cracking was observed in the additional CRDM nozzles inspected. RVH nozzle #32 was repaired using the repair technique described in CR-3 Relief Request 01-0002-RR. CR3 is developing a long-term strategy to deal with the CRDM nozzle-cracking issue. No previous similar CR-3 occurrences have been reported to the NRC.

1. FACILITY NAME	2. DOCKET	6. LER NUMBER			3. PAGE
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	
CRYSTAL RIVER UNIT 3	05000-302	01	- 004	- 00	2 OF 9

17. NARRATIVE *(If more space is required, use additional copies of NRC Form 366A)*

BACKGROUND

There are 69 Control Rod Drive Mechanism (CRDM) nozzles [AB, NZL] that penetrate the Reactor Vessel Head (RVH) [AB, RCT] (see Figure 1). The CRDM nozzles are approximately 5 feet long and are welded to the RVH at various radial locations from the centerline of the RVH. The nozzles are constructed from 4 inch outside diameter (OD) alloy 600 material. The lower end of the nozzle extends about 6 inches below the inside of the RVH.

The alloy 600 used in the fabrication of the CRDM nozzles was procured in accordance with the requirements of Specification SB-167, Section II to the 1965 Edition including Addenda through Summer 1967 of the American Society of Mechanical Engineers (ASME) Boiler & Pressure Vessel Code. The product form is tubing and the material manufacturer for the CRDM nozzles was the Babcock and Wilcox (B&W) Tubular Products Division. The 69 CRDM nozzles were manufactured from the same heat of material.

Each nozzle was machined to final dimensions to assure a match between the RVH bore and the OD of each nozzle. The nozzles were shrink fit by cooling and inserted in the closure head penetration, and then allowed to warm to room temperature. The CRDM nozzles were tack welded, then permanently welded to the closure head using alloy 182-weld metal. The manual, shielded metal arc welding process was used for both the tack weld and the J-groove weld. During weld buildup, the weld was ground and examined using liquid dye penetrant testing (PT) techniques incrementally through the thickness of the weld. The final weld was ground and PT inspected.

EVENT DESCRIPTION

At 1300, on October 1, 2001, Florida Power Corporation's (FPC's) Crystal River Unit 3 (CR-3) was in MODE 5 (COLD SHUTDOWN) at 0 percent RATED THERMAL POWER. While performing a qualified[1] visual inspection of the RVH, FPC personnel identified one potential leaking CRDM nozzle (nozzle #32), as evidenced by a buildup of boric acid crystals at the intersection of the nozzle with the RVH. No other CRDM nozzles inspected revealed a similar buildup. The RVH inspection was performed as part of Generic Letter 88-05, "Boric Acid Corrosion of Carbon Steel Reactor Pressure Boundary Components in PWR Plants," program and to satisfy a commitment made by FPC in response to NRC Bulletin 2001-01, "Circumferential Cracking of Reactor Pressure Vessel Head Penetration Nozzles."

Note [1]
Although Crystal River Unit 3 was committed to the performance of an effective visual examination during Refueling Outage 12, the examination performed met the intent of a qualified visual examination as described in NRC Bulletin 2001-01. This conclusion is based on evidence of boric acid crystal deposits found during the visual inspection which supports the results of an Engineering Evaluation performed using plant specific as-built CRDM nozzle interference fit data which concluded that through-wall pressure boundary leakage will produce visible boric acid crystal deposits on top of the RVH.

17. NARRATIVE *(If more space is required, use additional copies of NRC Form 366A)*

The amount of boric acid around nozzle #32 was estimated to be less than one cubic inch. After the RVH was removed and placed on the storage stand, additional inspections were performed.

At approximately 1748, on October 8, 2001, CR-3 was in MODE 6 (REFUELING) at 0 percent RATED THERMAL POWER. Ultrasonic testing (UT) examination performed on nozzle #32 revealed two through-wall axial cracks extending from the bottom of the nozzle, terminating above the J-groove weld. These cracks confirm that the boron deposits around CRDM nozzle #32 were indicative of a Reactor Coolant System (RCS) [AB] pressure boundary leak. This condition is consistent with industry experience with primary water stress corrosion cracking (PWSCC) in similar components that have been evaluated as part of Generic Letter 97-01, "Degradation of Control Rod Drive Mechanism Nozzle and Other Vessel Closure Head Penetrations," and NRC Bulletin 2001-01. The results of the UT examination of CRDM nozzle #32 are provided in the table below:

Crystal River Unit 3
CRDM Nozzle #32 (F12)*

Flaw #	Axial Location[1]			Circ. Location[2]			Remaining Ligament (From ID Surface)	Surface (ID/OD)
	Min. (in.)	Max. (in.)	Length (in.)	Min. (Deg)	Max. (Deg)	Extent (Deg)		
1	36.51			347.0	17.82	30.82	0.15 in.	OD
2	35.24	37.27	2.03	-132.25	63.14	195.39	Through-wall	OD
3	36.21	39.65	3.44	114.14			Through-wall	OD
4	36.41	39.65	3.24	170.18			Through-wall	OD
5	36.27			102.65	193.88	91.23	0.33 in.	OD

Notes:
* - Core position provided for convenience
1 - End of Nozzle @ 39.65 inches (Top of Nozzle Flange is Zero)
2 - Downhill Side of Nozzle @ 154.92 degrees
ID – inside diameter

As indicated in the above table, the UT data indicated the presence of five recordable indications including two (2) axially oriented cracks (flaw 3 and 4) that were through-wall, and extended from the bottom of the nozzle through and above the J-groove weld. These cracks originated at the weld-to-nozzle interface, propagated downward to the end of the nozzle, and upward through the weld into the annular space between the nozzle and the head. These two axial cracks were the primary source of leakage. These two cracks then joined a circumferential crack (flaw 5) on the OD of the nozzle above the weld. The circumferential crack (flaw 5) above the weld extended about 90 degrees and was approximately 50% through-wall. The UT identified one circumferential crack (flaw 1) below the weld. Flaw 1 extended for about 30 degrees and was within 0.15 inch of the inner diameter (ID) (approximately 75% through-wall).

17. NARRATIVE *(If more space is required, use additional copies of NRC Form 366A)*

Flaw 2 extended for about 195 degrees and was through-wall. Note that flaw 2 had both axial and circumferential characteristics, extending from below the weld, through the weld and above the weld. The largest portion of the flaw was below the weld (approximately 130 degrees). The crack at the deepest point was through-wall below the weld. However, this crack would provide a leak path to the annulus above the weld. All five cracks were OD initiated. No PT examination of the J-groove weld was required since through-wall cracking of the nozzle base material was confirmed.

The leakage of primary coolant through the CRDM nozzles was minimal and the leakage was detectable only by the observed accumulation of boric acid crystals at the intersection of the nozzle with the vessel head. However, Improved Technical Specification (ITS) Limiting Condition for Operation 3.4.12.a, limits Reactor Coolant System operational leakage to "No pressure boundary LEAKAGE" in MODES 1 through 4. Based on the above information, RCS operational leakage existed while at power.

At 1748, on October 8, 2001, a non-emergency eight-hour notification was made to the NRC Operations Center (NRC Event #38365) in accordance with 10 CFR 50.72(b)(3)(ii)(A) as any event or condition that results in a principal safety barrier being seriously degraded. This condition is also being reported pursuant to 10CFR50.73(a)(2)(i)(B) as a condition prohibited by Technical Specifications and pursuant to 10CFR50.73(a)(2)(ii)(A) as any event or condition that results in a principal safety barrier being seriously degraded.

CAUSE

The cause for the cracks observed in CRDM nozzle #32 is due to PWSCC. This conclusion is based on previous material, fabrication, and operating evaluations that have been performed, the results of the visual and ultrasonic examinations, and comparison of the flaws in nozzle #32 to similar flaws at other units. Alloy 600 and alloy 182 materials are susceptible to PWSCC.

A contributing factor is the grinding operation on the OD of the nozzles used to achieve the final interference fit in the RVH during the manufacturing process. This practice is now known to introduce a shallow cold work layer on the nozzle that creates higher residual stresses in the OD surface of the nozzle. This makes the OD surface more susceptible to crack initiation when exposed to a primary water environment, time and temperature.

SAFETY CONSEQUENCES

Framatome Report 51-5011603-01, "RV Head Nozzle and Weld Safety Assessment," evaluated the safety significance of CRDM PWSCC and concluded that it is not a significant safety issue, given that adequate visual examinations are conducted during refueling outages. Factors including structural margins, flaw propagation into the low alloy steel, head wastage due to boron leakage, and loose parts, were considered and found to be acceptable. A comparison of as-found conditions at CR-3 to the above factors indicates the following:

1. FACILITY NAME	2. DOCKET	6. LER NUMBER			3. PAGE
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	
CRYSTAL RIVER UNIT 3	05000-302	01	- 004 -	00	5 OF 9

17. NARRATIVE *(If more space is required, use additional copies of NRC Form 366A)*

1. Per EPRI Report TP-1001491, "PWR Materials Reliability Project Interim Alloy 600 Safety Assessments for U.S. PWR Plants (MRP-44), Part 2: Reactor Vessel Top Head Penetrations," May, 2001, the maximum permissible size for a through-wall circumferential flaw above the weld, while maintaining the ASME Code 3 times design pressure load limit, is approximately 273 degrees. The extent of the circumferential crack above the weld at CR-3 was approximately 92 degrees and was approximately 50 percent through-wall. This is significantly less than the calculated limit of 273 degrees. Therefore, significant structural margin remained in CRDM nozzle #32

2. No discernable RVH wastage was noted in the vicinity of the CRDM nozzle.

3. Based on evidence to date, Framatome Report 51-5011603-01 concludes that any significant cracking below the weld that could lead to a loose part would also be accompanied by through-wall cracking that would be detectable. Even if a loose part were to be generated and cause a control rod to stick, CR-3 Final Safety Analysis Report, Chapter 14.1.2.7, "Stuck-Out, Stuck-In or Dropped Rod Accident," fully evaluated the consequences of the highest worth control rod being stuck.

Based on the above factors, FPC concludes that the potential for cracking and leakage of a CRDM nozzle does not represent a reduction in the public health and safety. This event does not meet the definition of a Safety System Functional Failure

CORRECTIVE ACTIONS

Eight (8) CRDM's that were removed to support CRDM maintenance activities were inspected using UT. The locations of the eight additional nozzle inspections were 8 (L2), 21 (G5), 40 (G13), 52 (E13), 54 (O11), 58 (B8), 63 (B10), and 64 (F14). The results of the additional UT's indicated that there were no recordable indications in the eight CRDM nozzles inspected.

Nozzle #32 was repaired using the ambient temperature temper bead repair technique as described in CR-3 Relief Request 01-0002-RR. The final weld was examined using liquid dye penetrant and UT. No recordable indications were found. An in-service leakage test was performed in accordance with plant procedures. The plant conditions were normal operating pressure and temperature. No evidence of leakage was noted following a four-hour hold. Operability of the CRDM in nozzle #32 was confirmed during plant start-up in accordance with existing plant procedures.

CR-3 is developing a long-term strategy to deal with the CRDM nozzle-cracking issue. This corrective action is being addressed via the CR-3 Corrective Action Program.

1. FACILITY NAME	2. DOCKET	6. LER NUMBER			3. PAGE
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	
CRYSTAL RIVER UNIT 3	05000-302	01	- 004 -	00	6 OF 9

17. NARRATIVE *(If more space is required, use additional copies of NRC Form 366A)*

PREVIOUS SIMILAR EVENTS

No previous similar events at CR-3 have been identified. However, the nozzle cracking experienced on CRDM nozzle #32 is similar to the cracking experienced at Oconee Units 1, 2, 3, and Arkansas Nuclear One. This type of cracking has been experienced worldwide in the nuclear industry.

ATTACHMENTS

Figure 1 – Reactor Vessel Closure Head Map
Attachment 1 – Abbreviations, Definitions, and Acronyms
Attachment 2 – List of Commitments

NRC FORM 366A
(1-2001)

U.S. NUCLEAR REGULATORY COMMISSION

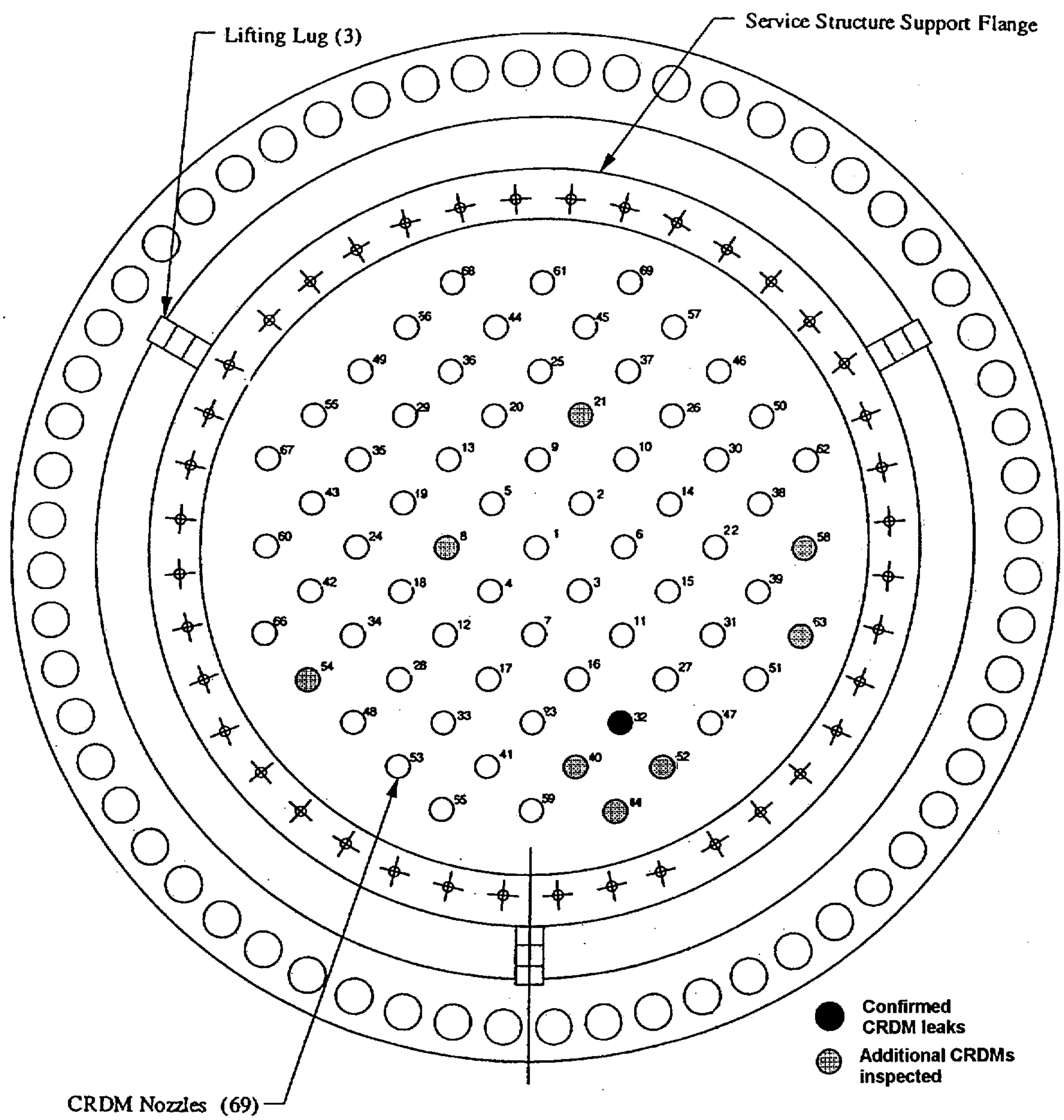
LICENSEE EVENT REPORT (LER)

1. FACILITY NAME	2. DOCKET	6. LER NUMBER			3. PAGE
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	
CRYSTAL RIVER UNIT 3	05000-302	01	- 004 -	00	7 OF 9

17. NARRATIVE *(If more space is required, use additional copies of NRC Form 366A)*

Figure 1

Reactor Vessel Closure Head Map



17. NARRATIVE *(If more space is required, use additional copies of NRC Form 366A)*

ATTACHMENT 1

ABBREVIATIONS, DEFINITIONS AND ACRONYMS

10CFR	Title 10 of the Code of Federal Regulations
ASME	American Society of Mechanical Engineers
B&W	Babcock and Wilcox
CRDM	Control Rod Drive Mechanism
CR-3	Crystal River Unit 3
FPC	Florida Power Corporation
ITS	Improved Technical Specifications
ID	Inside Diameter
LER	Licensee Event Report
OD	Outside Diameter
PT	Liquid Dye Penetrant Test
PWSCC	Primary Water Stress Corrosion Cracking
RCS	Reactor Coolant System
RVH	Reactor Vessel Head
UT	Ultrasonic Test

NOTES: Improved Technical Specifications defined terms appear capitalized in LER text {e.g., MODE 1}

Defined terms/acronyms/abbreviations appear in parenthesis when first used {e.g., Reactor Building (RB)}.

Energy Industry Identification System (EIIS) codes appear in square brackets {e.g., reactor building penetration [NH, PEN]}.

LICENSEE EVENT REPORT (LER)

1. FACILITY NAME	2. DOCKET	6. LER NUMBER			3. PAGE
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	
CRYSTAL RIVER UNIT 3	05000-302	01	- 004	- 00	9 OF 9

17. NARRATIVE *(If more space is required, use additional copies of NRC Form 366A)*

ATTACHMENT 2

LIST OF COMMITMENTS

The following table identifies those actions committed to by Florida Power Corporation in this document. Any other actions discussed in the submittal represent intended or planned actions by Florida Power Corporation. They are described to the NRC for the NRC's information and are not regulatory commitments. Please notify the Supervisor, Licensing & Regulatory Programs of any questions regarding this document or any associated regulatory commitments.

RESPONSE SECTION	COMMITMENT	DUE DATE
	No regulatory commitments are made in this submittal.	