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April 4, 2002

United States Nuclear Regulatory Commission
Document Control Desk
Washington, DC 20555

Operating Licenses DPR-58
Docket Nos. 50-315

Document Control Manager:

In accordance with the criteria established by 10 CFR 50.73 entitled Licensee Event Report System, the following report is being submitted:

LER 315/1999-007-01: "The Divider Between Upper and Lower Containment Volumes May be Overstressed" - Retraction.

No new commitments are identified in this submittal.

Should you have any questions regarding this correspondence, please contact Mr. Gordon P. Arent , Manager, Regulatory Affairs, at 616/697-5553.

Sincerely,

Joseph E. Pollock
Site Vice President

IJ/pae

Attachment

c: G. P. Arent
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IE22

NRC Form 366 (7-2001) U.S. NUCLEAR REGULATORY COMMISSION

LICENSEE EVENT REPORT (LER)

(See reverse for required number of digits/characters for each block)

APPROVED BY OMB NO. 3150-0104 EXPIRES 7-31-2004

1. FACILITY NAME	2. DOCKET NUMBER	3. PAGE
Donald C. Cook Nuclear Plant Unit 1	05000-315	1 of 3

4. TITLE

Retraction - The Divider Barrier Between Upper and Lower Containment Volumes May Be Overstressed

5. EVENT DATE			6. LER NUMBER				7. REPORT DATE			8. OTHER FACILITIES INVOLVED	
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER		REVISION NUMBER	MONTH	DAY	YEAR	FACILITY NAME	DOCKET NUMBER
10	20	1998	1999	--	007	-- 01	04	04	2002	FACILITY NAME D. C. Cook Unit 2	DOCKET NUMBER 05000-316

9. OPERATING MODE	5
10. POWER LEVEL	00

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)		50.36(c)(1)(i)(A)		50.73(a)(2)(iv)(A)		73.71(a)(4)
	20.2203(a)(2)(i)		50.36(c)(1)(ii)(A)		50.73(a)(2)(v)(A)		73.71(a)(5)
	20.2203(a)(2)(ii)		50.36(c)(2)		50.73(a)(2)(v)(B)	X	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)		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)		50.73(a)(2)(ii)(A)		50.73(a)(2)(viii)(B)		

12. LICENSEE CONTACT FOR THIS LER

NAME	TELEPHONE NUMBER (Include Area Code)
I. N. Jackiw, Regulatory Affairs	(616) 465-5901, X1602

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

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO EPIX	CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO EPIX

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)

This LER is being retracted.

On October 20, 1998, based on preliminary stress analyses, engineering personnel determined that a postulated steam line break could result in calculated stresses exceeding the code allowable stresses for the Unit 1 Steam Generator (SG) enclosures. These reinforced concrete enclosures are part of the divider barrier between the upper and lower containment volumes. Overstressed conditions in these enclosures could allow increased steam bypass flow around the ice condensers, which could result in higher than expected containment pressure. Due to a lack of reasonable assurance that the structures would not be overstressed, on March 2, 1999, at 1314 hours, an Emergency Notification System (ENS) notification was made in accordance with 10CFR50.72(b)(2)(i), for a condition outside the design bases of the plant. An LER was submitted on April 1, 1999, in accordance with the related 10CFR50.73(a)(2)(ii) requirement.

When the ENS notification was made, Donald C. Cook Nuclear Plant (CNP) was applying the conservatisms of two different stress analysis methods into a combined stress limit, creating a third method of analysis. This third method conservatively bounded the licensing requirements, but resulted in preliminary calculated overstresses of the SG subcompartment structures. CNP performed an in-depth review of the licensing basis and determined that the two different stress analysis methods are individually adequate, and need not be combined. The potential overstresses were calculated using methods beyond the CNP licensing bases.

An overstress condition for the load combinations applicable to the steam generator enclosure analyses did not exist for the pressure values calculated at the time of licensing, and LER 50-315/1999-007-00 is hereby retracted.

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

Conditions Prior to Event
Unit 1 – Shutdown
Unit 2 – Shutdown

Description of Event

On October 20, 1998, based on preliminary stress analyses, engineering personnel determined that a postulated steam line break could result in calculated stresses exceeding the code allowable stresses for the Unit 1 steam generator (SG) enclosures. These reinforced concrete enclosures are part of the divider barrier between the upper and lower containment volumes. Overstressed conditions in these enclosures could allow increased steam bypass flow around the ice condensers, resulting in higher than expected containment pressure.

When the potential overstress conditions were identified, it was initially concluded that the conditions could be due to significant conservatisms used in the preliminary calculation and in its interpretation. However, as engineering evaluations continued, a lack of information confirming an acceptable condition led to the conclusion that there was no longer a reasonable expectation that the structures would not be overstressed. On March 2, 1999, at 1314 hours, an Emergency Notification System (ENS) notification was made in accordance with 10 CFR 50.72(b)(2)(i), for a condition outside the design bases of the plant. An LER was submitted on April 1, 1999, in accordance with the related 10 CFR 50.73(a)(2)(ii) requirement.

Basis for Retraction

At the time the containment structures were designed (early 1970's), Donald C. Cook Nuclear Plant (CNP) did not precisely define the SG subcompartment concrete structure pressure load distribution resulting from a main steam line break (MSLB). Instead, a simplified pressure load definition was used. The simplified pressure load was defined as 20 pounds per square inch differential (psid) applied uniformly. Simplifications and uncertainties inherent in this definition were addressed by applying multipliers (i.e., uncertainty factors) of up to 1.5 to the simplified pressure load when evaluating various load combinations. These load combinations were then defined in Chapter 5 of the Updated Final Safety Analysis Report (UFSAR).

Between 1977 and 1978, available analytical methods were used to better define the SG enclosure pressure load distribution by evaluating nodalized subcompartment pressure distribution. Uncertainty factors (multipliers) are no longer applied to the pressure parameter. Instead, a unity load factor is used and the effects of the combined loads are compared to the structure's capability. A 1.5 factor above the ultimate capacity for the SG enclosure was shown. This load combination was then defined in Chapter 14 of the UFSAR.

During the initial licensing of CNP, both analysis methods were used and remain as current CNP licensing requirements. UFSAR Chapter 5 requires the factored load method, and the UFSAR Chapter 14 requires a stratified pressure distribution. These two requirements are separate methods of analysis, each with their own engineering conservatisms. The UFSAR requires that both methods be used in calculating acceptable SG enclosure structure stresses.

At the start of the Unit 1 SG generator replacement project, the decision was made to bound the analyses by applying the load factors detailed in UFSAR Chapter 5 to the more precisely defined subcompartment pressures found in Chapter 14. As a result, the pressure acting on the critical SG subcompartment roof sections was first increased from 20 psid to 34 psid, and then multiplied by a 1.5 uncertainty factor. The resultant calculated pressure load acting on the critical roof sections was determined to be substantially greater than that required by either UFSAR Chapter 5 or Chapter 14. This decision was responsible for the preliminary conclusion that the Unit 1 SG enclosures would be overstressed.

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

When the ENS notification was made in 1998, CNP was applying the conservatisms of both USFAR Chapters 5 and 14 methods into a combined stress limit. Subsequently, CNP performed an in-depth review of the licensing basis to specifically identify the differences or similarities between the load combinations specified in UFSAR Chapters 5 and 14. This review determined that the analytical methods described in both USFAR chapters were individually adequate, and need not be combined.

The potential overstresses that were calculated by combining the conservatisms of both Chapters 5 and 14 methodologies, were not required to be analyzed because they were calculated using methods beyond CNPs licensing basis. During a public meeting on June 11, 2001, CNP presented information related to the design and licensing basis of containment structures which were regarded as degraded but operable since the restart of both units at D. C. Cook. CNP concluded that all calculations concerning the containment structures including the SG enclosures had been completed and the analyses confirmed that all containment structures have an adequate margin and comply with the design basis requirements as stated in the D. C. Cook Updated Final Safety Analysis Report.

Conclusion

The ENS notification for this condition was made without fully realizing the licensing basis of CNP. It became clear that an overstress condition for the load combinations applicable to the steam generator enclosure analyses did not exist for the pressure values calculated at the time CNP was licensed. As such, LER 50-315/1999-007-00 is hereby being retracted.

Previous Similar Events

LER 316/2000-003-01, "Containment Internal Concrete Structures Do Not Meet Design Load Margins," dated 11-20-00 identified that using a revised Nuclear Steam Supply System (NSSS) vendor transient mass distribution (TMD) containment analysis, some Unit 1 and 2 containment internal concrete sub-compartment structural elements did not meet the design pressure load factor margin of 1.5 as described in the CNP Updated Final Safety Analysis Report (UFSAR). Actions undertaken to resolve these containment structural issues included reconstituting the existing TMD analyses with new input parameters and then using the new TMD analyses results to complete conservative simplified containment structural evaluations. CNP concluded that all calculations concerning the containment structures had been completed and the analyses confirmed that all containment structures have an adequate margin and comply with the design basis requirements as stated in the D. C. Cook Updated Final Safety Analysis Report. Final Closure Report, NED-2001-069-REP revision 0, provides the resolution and closure of the CNP containment structural issues. These issues were discussed during numerous public meetings conducted in 2000 and 2001.