UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

January 3, 1991

Mr. Edward Griffing
Manager, Technical Division
Nuclear Management and Resources Council
1776 Eye Street, N.W.
Suite 300
Washington, D.C. 20006-2496

Dear Mr. Griffing

SUBJECT: PRESSURE WATER REACTOR VESSEL LICENSE RENEWAL INDUSTRY REPORT

By letter dated December 26, 1990, NUMARC submitted responses to the NRC comments on the subject Industry Report (IR), which were transmitted to NUMARC by letter dated September 14, 1990. A meeting to discuss these responses has been scheduled for February 7, 1991 as requested by NUMARC.

Enclosed are additional NRC staff comments on the subject IR. We understand that NUMARC would have limited time to address these additional comments before the scheduled meeting, but request that you be prepared to discuss NUMARC's responses to these additional comments, to the extent possible, during the scheduled meeting.

The additional staff comments consist of 18 specific comments. We believe that 7 of these comments have been addressed, in whole or in part, in NUMARC's responses dated December 26, 1990. The following is the cross reference for these 7 comments, based on the additional comment number and the NRC comment number in the December 26, 1990 letter:

Additional Comment Number	Previous NRC Comment Number
1	3
2	2,7
6	16
7	8
11	2,7
14	10
15	11

NUMARC has the option to provide additional discussions relating to these 7 additional comments. Further, there are 5 additional comments relating to fatigue, specifically, additional comment numbers 3, 12, 16, 17, and 18, and these will be resolved on a generic basis at a later date. The remaining 6 additional comments, specifically, additional comment numbers 4, 5, 8, 9, 10, and 13, should be addressed by NUMARC.

DF93 ~~DF~~

Ofm-7 '\'
NUMARC
X RD-10-1
PWR

new RIDS code, per 1/10/91 telcon w/J. mcknight. KHV.

If you have any questions concerning this matter, please contact either myself or P. T. Kuo at 492-3147.

Sincerely,

/s/

John W. Craig, Director
License Renewal Project Directorate
Division of Advanced Reactors
and Special Projects
Office of Nuclear Reactor Regulation

Enclosure:
As stated

cc: K. Cozens, NUMARC

Distribution:
Central Files
J. Partlow
W. Minners
J. Durr, RI
LRPD r/f
R. Kirkwood
D. Crutchfield
F. Gillespie
A. Thadani
C. Y. Cheng
J. Norberg
L. Shao
G. Bagchi
R. Bosnak
J. Vora
J. Richardson
M. Mayfield
C. McCracken
R. Jones
W. Travers
R. Borchardt
PDR

LRPD SSL	LRPD:SC SSL	LRPD:D JC
SLee	for PTKuo	JCraig
01/03/91	01/03/91	01/03/91

OFFICIAL RECORD COPY

If you have any questions concerning this matter, please contact either myself or P. T. Kuo at 492-3147.

Sincerely,

John W. Craig, Director
License Renewal Project Directorate
Division of Advanced Reactors
and Special Projects
Office of Nuclear Reactor Regulation

Enclosure:
As stated

cc: K. Cozens, NUMARC

If you have any questions concerning this matter, please contact either myself or P. T. Kuo at 492-3147.

Sincerely,

/s/

John W. Craig, Director
License Renewal Project Directorate
Division of Advanced Reactors
and Special Projects
Office of Nuclear Reactor Regulation

Enclosure:
As stated

cc: K. Cozens, NUMARC

Distribution:

Central Files	D. Crutchfield	G. Bagchi	C. McCracken
J. Partlow	F. Gillespie	R. Bosnak	R. Jones
W. Minners	A. Thadani	J. Vora	W. Travers
J. Durr, RI	C. Y. Cheng	J. Richardson	R. Borchardt
LRPD r/f	J. Norberg	M. Mayfield	PDR
R. Kirkwood	L. Shao		

LRPD SSL	LRPD:SC SSL	LRPD:D JC
SLee	for PTKuo	JCraig
01/03/91	01/03/91	01/03/91

OFFICIAL RECORD COPY

PRESSURE WATER REACTOR VESSEL LICENSE RENEWAL INDUSTRY REPORT

SPECIFIC COMMENTS

(1) p. 1-5, para. 1.3.2 - Under Neutron Irradiation Embrittlement, the IR implies that vessel components in the beltline region are subject to radiation embrittlement degradation but that the IR will only address the intermediate and lower shells. The IR should also address the beltline regions welds and the fact that some larger vessels have 3 shells in the beltline region.

(2) p. 1-8, para. 1.3.2 - Under Stress Corrosion Cracking, CRDM housings and instrumentation tubes are listed as vessel components not subject to SCC as a significant age-related degradation mechanism. Since SCC was recently discovered to have occurred on similar components of the pressurizer of at least one US PWR plant, it is not clear that SCC for CRDM housings and instrumentation tubes should be arbitrarily classified as non-significant in the IR without providing the technical rationale.

(3) p. 1-14, FATIGUE - This section refers to fatigue as a degradation mechanism that is resolved. The IR does not provide an adequate basis for the conclusion. Reactor vessels are designed to the ASME Code assuming numbers of transient and operating cycles associated with the initial 40 year operating life. As noted in the IR, some plants are required by their Tech. Specs. to monitor their operating cycles and assure that the reactor vessel is operated within its fatigue design life. It is our understanding that the Tech. Spec. requirements only apply to more recently licensed, i.e., newer, plants. The IR notes that methods exist for monitoring of plant transients and recalculation of fatigue usage factors, but specifically does not indicate that this will be done for any specific vessel.

Additionally, the IR notes that Inservice Inspection in accordance with ASME Section XI will provide assurance that only significant fatigue crack growth will be detected. For the many older vessels that are not completely inspectable in accordance with Section XI, particularly in the region of the primary coolant nozzles, a high fatigue usage factor location, the IR does not provide a technical rationale for stating that the Fatigue degradation mechanism is resolved.

(4) p. 2-2, First para. - This paragraph lists the various PWR reactor vessel components that are considered addressed by the IR. We note that Core Flood Nozzles of B&W vessels, Safety Injection Nozzles of W 2 Loop Plant vessels, Upper Head Injection Nozzles in some W 4 Loop Plant vessels, and Closure Head Vents in all vessels are not listed. The IR should include all of these components.

(5) p. 3-19, Operating and Maintenance History - It is not clear what conclusions are to be drawn from this entire section. Various degradation mechanisms and operational problems are briefly discussed, but no statement or conclusion is provided as to their relationship to license renewal.

SPECIFIC COMMENTS (continued)

(6) p. 3-20, 21, 22 Section 3.3.2 - The discussion of underclad cracking does not adequately address the aging/life extension concerns. Cracks or flaws that were not removed were previously evaluated and found to be acceptable for the initial 40 year life for vessel base metal and cladding taking into account transient and operating cycles and material properties associated with the first 40 years of operation. Particularly for the core region of such vessels, have evaluations of underclad crack growth been performed taking into account the increased number of operating cycles and the effects of increased material embrittlement from higher radiation levels and thermal aging associated with life extension?

(7) pp 1-3, 1-8, 4-19 & -20, and 4-30 - The discussion in the IR on low-temperature thermal aging of ferritic steel in the vessel states unambiguously that there is no reason for concern. This is in contradiction to the studies at Harwell in the UK which have demonstrated that comparable shifts in the transition temperature can be caused by low-temperature thermal aging as by irradiation embrittlement in current generation low-copper steels. The British have gone so far as to include a shift in transition temperature of up to 30 F purely due to thermal aging in their planning for the Sizwell B reactor. While unlikely to be a grave concern, there are grounds to believe that a shift in the transition temperature of the vessel wall can be attributed to such a mechanism and should therefore be included in life extension considerations. Of particular concern should be the consideration of such a shift on the nozzles and other components of the vessel, which are generally not subject to irradiation-induced embrittlement, to ascertain if their stress concentrations, in conjunction with a shifted transition temperature, could lead to unexpected failure.

(8) Section 4.4 - In addition to the concerns about low-temperature thermal aging of the ferritic steel, it is our impression that at least in some plants the CRDM housings are cast stainless steels and not wrought product as identified in the report. Although this is a plant specific feature, the possibility that it may have to be considered should be in the IR.

(9) p. 4-6, Section 4.1.3; p. 4-14, Section 4.3.1; p. 4-19, Section 4.4.1 - In these three sections any consideration of degradation of the stainless steel cladding due to irradiation, SCC, or thermal aging is dismissed. For irradiation and thermal aging, the argument is made that no credit is taken for cladding in the analyses and therefore the degradation mechanisms are not a factor. For SCC, it is argued that the combination of a resistant material and a non-aggressive environment eliminate this mechanism for cladding. These conclusions cannot be supported, as discussed below.

The argument that no credit is taken for cladding is only partly true. It is implicitly assumed that the cladding is a tough material and that a conservative approach would be to assume that this material has the same toughness as the ferritic steel. However, research in recent years has

SPECIFIC COMMENTS (continued)

shown that the toughness of irradiated cladding is no better than the toughness of the so-called low Charpy upper shelf energy welds -- a material of great concern today. Thus, for many plants, the tough base metal is clad with a material that has a lower fracture toughness, and this may lead to analyses that are less conservative than anticipated. This situation will only be exacerbated by thermal aging.

The argument that cladding is resistant to SCC also cannot be supported. It is noted in comment (10) that current practice in BWRs is to characterize the environment in terms of the electrochemical potential rather than simply the dissolved oxygen content. Thus, it is not clear that the PWR environment cannot promote SCC in a susceptible material. The argument that cladding is not a susceptible material because it is required to have a ferrite content greater than 5% is only true for newer plants, and then may not be true near weld repairs. Thus, the basis for the conclusion that cladding is not susceptible to SCC may not be valid for any plant, and certainly is not valid generically.

(10) p. 4-13 - Tests in Sweden have shown that it is difficult to identify a safe level of dissolved oxygen to assure that SCC of sensitized stainless steels cannot occur. Current practice in BWRs does not rely on achievement of a particular oxygen level but rather focuses on electrochemical potential (ECP) to characterize the environment. No such characterization of the PWR is cited. What assurance can be provided that the plant specific environment will not promote SCC?

(11) The report appears to understate the potential problems associated with PWSCC of Alloy 600. As the review in Section 3.3 notes, although it occurs most frequently in steam generator tubing, other cracking incidents have occurred. Every welded or cold worked Alloy 600 has a potential for high residual stresses (service loads in components or steam generator tubing by themselves would be unlikely to cause cracking), and the post-weld heat treatment given the ferritic vessel does little to relieve residual stresses in Alloy 600. Experience has shown that this is a susceptible material in a susceptible environment. Since the stress levels in these components (applied plus residual) are unknown, the possibility of PWSCC cannot be ruled out.

(12) p. 4-10, Section 4.2.2 - The point is made that the assessment of potential fatigue damage in this IR applies only to Section III vessels. Thus, the IR does not address fatigue considerations for the older PWRs that were designed and built to the rules of ASME Section VIII. This issue must be addressed either in the IR or in separate documents prepared by the industry or by the NRC.

SPECIFIC COMMENTS (continued)

(13) p. 4-15 - The report states "there is no conclusive evidence...that SCC of low alloy steel is a concern in a PWR reactor environment" and then concludes that SCC of low alloy steels is not a significant mechanism of degradation. However, the question is whether there is conclusive evidence that it is not a concern. Although most available evidence suggests that even for high sulfur steels, SCC is unlikely in materials exposed to a "typical" reactor environments, this may not necessarily be true under crevice conditions, and the possibility of SCC in such regions needs to be considered further.

(14) p. 4-28, Section 4.8, Mechanical Wear - The treatment of mechanical wear is not adequate. For example, two wear mechanisms are loose parts and movement of thermal shields. An example of wear occurred at Yankee Rowe when a surveillance capsule broke loose and battered a hole through the fingerprint cladding. This section could be expanded and made stronger. An obvious solution is the presence of a loose parts monitor; however, monitors aren't that common on reactors.

Another mechanism that is not discussed is the fretting -- corrosion? erosion? wear? -- of the instrument tubes. This is a problem for Westinghouse vessels and led to NRC Information Notices 87-44 and 88-09. Information Notice 88-09 requires an inspection program and "the establishment, with technical justification, of an appropriate inspection frequency." Since this degradation mechanism is time dependent, it is a problem that must be addressed for license renewal. The IR should be modified to address this degradation mechanism for instrument tubes.

(15) p. 5-15, Section 5.1.5 - The IR states that all issues related to neutron irradiation embrittlement are adequately addressed by current practices and then dismisses neutron irradiation embrittlement of the reactor pressure vessel. The statement is not correct. The surveillance program is not adequately addressed by the current ASTM E 185 or by 10 CFR 50 App. H. The surveillance programs must be revised to address the renewal period. An obvious activity that must be started now to address this issue would be to reconstitute the Charpy specimens from the existing surveillance packages. The IR must be modified to address this issue.

(16) p. 5-15, Section 5.2, Fatigue - The discussion of fatigue in the report is based on the assumption that the fatigue design rules of ASME Section III and the crack growth technology in Section XI are inherently conservative. The current Section XI crack growth curves are already known to be nonconservative under some conditions even for simple, well-defined loading histories. These curves are expected to be revised shortly to better reflect crack growth rates obtained for laboratory specimens under simple loading histories, but even then little data is available to assess the degree of conservatism of the technology when dealing with realistic crack geometries, residual stresses, and complex loading histories. Similarly, the fatigue design rules of Section III are definitely not conservative for carbon steels in BWR environments. Environmental effects appear to be less significant in PWR environments, but the report should at least address the question. The possible effect

of the environment on fatigue life isn't even mentioned in the report. The report should be modified to include a competent treatment of fatigue as a damage mechanism and how it relates to various components of a reactor pressure vessel.

(17) p. 5-15, Section 5.2.1 - The intent of the first sentence is not clear. The sentence states that fatigue damage accumulation is "tracked" to provide assurance that fatigue does not cause cracks to initiate in reactor vessel components. It is the staff's understanding that operating cycles and their effect on reactor vessel fatigue accumulation are not tracked at many older plants. This section of the report should be expanded to describe the method that is to be used to resolve the fatigue degradation mechanism concern for those plants that have not been monitoring the effect of plant operating cycles on fatigue usage factor during the first 40 years of vessel operation.

(18) p. 5-16, second para. - This paragraph states that "plant Technical Specifications require that major design cycles be tracked during the design operating life in order to assure the satisfaction of fatigue limits." As stated previously, it is the staff's understanding that the Technical Specifications at many older operating plants do not contain this requirement. The IR should be expanded to address the resolution of the fatigue degradation mechanism concern for these plants.