UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

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BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of	)	
	)	Dockets Nos. 50-250 OLA-4
FLORIDA POWER & LIGHT COMPANY	)	50-251 OLA-4
	)	
(Turkey Point Plant,	)	(Pressure/Temperature Limits)
Units 3 and 4)	)	
	)	

INTERVENORS' RESPONSE TO
LICENSEE'S MOTION FOR SUMMARY
DISPOSITION OF INTERVENORS' CONTENTIONS

Pursuant to 10 C.F.R. 2.749, Intervenors, the Center for Nuclear Responsibility and Joette Lorion (Intervenors), hereby file their response to Licensee's motion for summary disposition in the above captioned proceeding. In support of this response, Intervenors have attached "Intervenors Statement of Material Facts As To Which There Is A Genuine Issue To Be Heard With Respect To Intervenors' Contentions" and the letter of Dr. George Sih on Contention 2 dated October 18, 1989 (Sih Letter, Attachment A). As discussed below, the Intervenors contend that there is a genuine issue of material fact regarding the matters set forth in the attached statement and affidavit, and that the Licensee is not entitled to a decision in its favor as a matter of law and summary judgment should be denied.

I. BACKGROUND OF THIS PROCEEDING

On October 19, 1988, a notice was published in the Federal Register announcing the proposed issuance of amendments to the Technical Specifications for Turkey Point Units 3 and 4. 53 Fed. Reg. 40988. The Proposed amendments would modify the pressure/temperature limits for the reactor coolant system and the pressurizer for each unit.

On Novemeber 17, 1988, the Center for Nuclear Responsibility, Inc. ("Center") and Joette Lorion, collectively referred to herein as "Intervenors", filed with the Nuclear Regulatory Commission ("NRC") a Request for Hearing and Petition for Leave to Intervene ("Petition") concerning the Florida Power & Light ("FPL") amendment request.

On January 10, 1989, the NRC Staff issued Amendment Nos. 134 and 128 to the operating licenses for Turkey Point, Units 3 and 4 respectively, revising the pressure/temperature ("P/T") limits for the Turkey Point units along with their Safety Evaluation and Final Determination of No Significant Hazards Consideration.

The Intervenors then submitted their "Amended Request for Hearing and Petition for Leave to Intervene" on February 17, 1989, which listed three Contentions that Intervenors asked to be admitted for litigation in this proceeding. On March 21, 1989, the Atomic Safety and Licensing Board (Board) held oral argument on the contentions. Subsequently, on June 8, 1989, the Board issued an Order which denied Contention 1 and accepted portions of Contentions

2 and 3.

On September 8, 1989, after a meeting with the Licensee, Intervenors withdrew Contention 3 from this proceeding. Finally, on September 11, 1989, the Licensee filed their Motion for Summary Disposition of Intervenor's Contentions.

II. LEGAL STANDARD FOR SUMMARY DISPOSITION

The summary disposition procedure should be utilized on issues where there is no genuine issue of material fact to be heard so that evidentiary hearing time is not wasted on such issues. Statement of Policy on Conduct of Licensing Proceedings, CLI-81-8, 13 NRC 452, 457 (1981); Wisconsin Electric Power Co. (Point Beach Nuclear Plant, Unit 1), ALAB-696, 16 NRC 1245, 1263 (1982); Houston Lighting and Power Co. (Allens Creek Nuclear Generating Station, Unit 1), ALAB-590, 11 NRC 542, 550 (1980).

It is the movant, not the opposing party, which has the burden of showing the absence of a genuine issue as to any material fact. Cleveland Electric Illuminating Co. (Perry Nuclear Power Plant, Units 1 and 2), ALAB-443, 6 NRC 741, 753 (1977). Since the moving party has the burden to show initially the absence of a genuine issue concerning any material fact, where the evidentiary matter in support of the motion does not establish the absence of a genuine issue, summary judgment must be denied even if no opposing evidentiary matter is presented. Adickes v. Kress & Co., 398 U.S. 144, 160 (1970). However, if the motion for summary disposition is properly supported, the oppsition may not rest upon "mere allegations or denials"; rather, the answer "must set forth specific facts showing that there is a genuine issue of fact." Virginia Electric and Power Co. (North Anna Power Station, Units 1 and 2), ALAB-584, 11 NRC 451, 453 (1980).

A. BACKGROUND

> There is a high, increasing likelihood that someday soon, during a seemingly minor malfunction at any of a dozen or more nuclear power plants around the United States, the steel vessel that houses the radioactive core is going to crack like a piece of glass. The result will be a core meltdown, the most serious kind of nuclear accident.
>
> Demetrios Basdekas, NRC Safety Engineer
> "The Risk of a Meltdown,"
> New York Times (March 29, 1982), (Exhibit 1).

Two facts have been known since our nation undertook the commercial development of nuclear power. 1) the absolute integrity of the large steel vessel tht houses the core and contains the cooling water for the reactor is central to protecting the health and safety of the adjoining community and the environment and 2) all metals, including steel, become embrittled overtime as a result of continued exposure of neutron irradiation. (Exhibit 2).

Nuclear plant pressure vessels are fabricated from ferritic steels. At Turkey Point, for instance, large sections of eight inch thick steel are welded together circumferentially to form the housing for the reactor core.

The safety of the public depends on the ability of the materials in the vessel and the welds to maintain their fracture toughness. Fracture toughness is a material property that enables the material to resist brittle fracture when stressed. An adequate level of fracture toughness provides the assurance that small flaws or cracks will not propagate in a "brittle manner" as a result of stresses caused by reactor heatup, cooldown and/or abnormal transients.

It is well known that for steels used in nuclear reactor pressure vessels and their welds, three considerations are important. First, fracture toughness increases with increasing temperatures; second, fracture toughness decreases with increasing load rates, and third, fracture toughness decreases with neutron irradiation.

In recognition of these considerations, power reactors are operated within restriction imposed by the Technical Specifications on the pressure during heatup and cooldown operations. These restrictions assure that the reactor vessel will not be subjected to that combination of pressure and temperature that could cause brittle fracture of the vessel if there were significant flaws in the vessel material. The effect of neutron radiation on the fracture toughness of the vessel material is accounted for in developing and revising these Technical Specification limitations over the life of the plant. The pressure/temperature limits, which are the subject of this proceeding are just such restrictions.

Additionally, there is another issue to consider where fracture toughness is concerned. That is the fact that in many of the older nuclear plants, such as Turkey Point, high levels of copper and nickel were used to fabricate the welds of the vessels and in some cases the vessels themselves. These elements were later shown to result in greater irradiation damage to the vessel material than had been initially expected. Irradiation damage in these plants caused a shift in the fracture toughness curve to higher temperatures, and therefore, increased the possibility of a nonductile failure.

This is so because as metal embrittles, it loses the property of "ductility" and must be kept at increasingly high temperatures in order to retain adequate ductility to avoid cracking or shattering in response to stresses or shocks. (Exhibit 2).

In 1981, the Nuclear Regulatory Commission became concerned about the extent of the embrittlement problem at some of the nation's older nuclear power reactors. This concern was manifested as a result of NRC Safety Engineer, Demetrios Basdekas' warnings that some of the more embrittled reactors with high copper contents in their welds could shatter from pressurized thermal shock (PTS) and endanger the communities in which the plants were located. (Exhibits 1 and 3). As part of the NRC's investigation of the (PTS) phenomenon, they sent letters pursuant to 10 C.F.R. 50.54 to Licensees whose fracture toughness of their reactor pressure vessels were approaching levels of concern. (Exhibit 4).

Florida Power and Light Company received just such a letter concerning the Turkey Point Unit 4 reactor. FPL was asked to submit plant specific information to the NRC in 150 days in lieu of licensing action. (id.). The Licensee was not asked to submit information on Unit 3, nor did the NRC single out Unit 3 as one of the nuclear power reactors that concerned them.

Yet, when the Licensee responded to the NRC's 50.54 letter on August 23, 1981, concerning a question the NRC had proposed as to the reference temperature nil-ductility transfer value (RTNDT) for Unit 4, the Licensee responded that the value they had provided the NRC was based on Unit 3 data which had been shown to be more representative of Unit 4 than the surveillance capsule that had been removed from Unit 4. (Exhibit 5).

The surveillance capsules that the Licensee was refering to were samples of weld material that they and other licensees are required to install in each reactor vessel so that they can be periodically withdrawn and tested to determine the actual extent of the embrittlement that has occurred in the specific reactor vessel. (Exhibit 2).

These samples are required by 10 C.F.R. Appendices G and H to be withdrawn periodically and subjected to a process known as "Charpy" tests. In these tests, specimens are heated to different temperatures and then struck to determine the temperature at which the metal shatters or cracks in order to determine the extent of embrittlement and the minimum temperature that must be maintained in order to assure the metal retains sufficient ductility to resist anticipated shocks (id.). The danger-point occurs at the temperature at which the metal loses its ductility (or arrives at "nil ductility"). The Commission and the industry use the term "reference temperature for nil ductility transition," abbreviated as "RT_{NDT}", to identify this danger-point.

In 1974 and 1975, the Licensee removed weld metal capsules T from Turkey Point Units 3 and 4 and Charpy tests were performed separately on the samples from each unit. (Exhibit 6 and 7).

The central document necessary to demonstrate the basis for Intervenors continuing concerns regarding Unit 4 is a report submitted by the Southwest Research Institute (the "Institute") entitled Pressure Temperature Limitation for the Turkey Point Unit Nos. 3 and 4 Nuclear Power Plants, SWRI Project No. 02-4383-039 (June 30, 1976). (Exhibit 8). The Institute had conducted Charpy tests on metals contained in a capsule taken from Unit No. 4 (Exhibit 7). Materials contained in the capsule taken from Unit No. 3 had been tested by the Westinghouse Electric Corporation (Exhibit 6). Thereafter, the Institute was asked to project the separate "heatup and cooldown limit curves" for the vessels for Units No. 3 and No. 4 applying the Commission's prescribed computational criteria to the separate test results on materials taken from each of the two units (id.). The Institute's summary of its results, set forth in the margin, illustrates the dramatic difference in embrittlement found in the Unit No. 3 samples from that found in the Unit No. 4 samples after less than three years operation (Exhibit 8). The data also suggests that, as early as 1976, the Commission and FPL were aware that the best available data indicated that the embrittlement occurring in Unit No. 4 would require that the temperature of that vessel be maintained at well-above 300 degrees F to maintain acceptable ductility before the Unit had been in operation for the equivalent of ten effective full power years ("EFPY") (id.).

The values of RT_{NDT} for the beltline regions of Turkey Point Units Nos. 3 and 4 were derived from (1) the surveillance program test results, (2) computed ratios of fast flux at the 1/4 and 3/4 locations in the vessel wall, and (3) trend curves in RT_{NDT} as a function of neutron fluence (E 1 MeV). A summary of these values is as follows:

Unit No.	Operating Period	RTNDT at 1/4 T	RTNDT at 3/4 T
3	5 EFPY	194 deg.F	131 deg.F
3	10 EFPY	236 deg.F	159 deg.F
4	5 EFPY	281 deg.F	188 deg.F
4	10 EFPY	342 deg.F	230 deg.F

* EFPY = Effective Full Power Year

E. Norris and J. Unruh, Pressure-Temperature Limitations for the Turkey Point Unit Nos. 3 & 4 Nuclear Power Plant at 27 (SWRI Project No. 02-4383-039 (June 30, 1976). (Exhibit 8).

How does a reactor whose pressure vessel that the Institute's 1976 report projected would exceed the NRC's own 300 deg. F screening criterion after less than ten Efective Full Power Years (EFPY) continue to operate ? The public record suggests that continued operation is the product of legal alchemy rather than technical progress. The legal alchemy was achieved simply and in a manner that would have been impossible if the NRC Staff had not allowed the Licensee to calculate the RTNDT for Unit 4 based upon "Unit 3 data" in response to the Commission's 1981 50.54 letter. (Exhibits 4 and 5).

Thus, it appears that the NRC Staff allowed the Licensee to use an integrated surveillance program to calculate the embrittlement of Unit 4 long before they confirmed the practice on April 22, 1985 when they issued a license amendment to FPL which allowed them to use an integrated surveillance program to calculate radiation damage to the Turkey Point reactor vessels.

As did the Licensee, the NRC appears to have ignored the actual differences in levels of embrittlement disclosed by the 1976 reports for Units 3 and 4, and has authorized FPL to continue operating Unit 4 so long as Turkey Point Unit 3 meets the Commission's embrittlement criterion. The record before this Board now suggests that the NRC Staff continues to ignore the fact that the only data ever derived from weld metal tests for Unit 4 demonstrates that it is non-conservative and improper to calculate the ART and revise the P/T limits for Unit 4 based primarily on data from the less severely affected Unit 3.

Intervenors contend that neither the Licensee nor the Staff have given the Board proper justification for thir decision not to test Unit 4's capsule V weld metal specimen in order to revise the P/T limits for that unit. A decision, which if sanctioned by this Board, could make a rupture of the reactor pressure vessel with its enormous public health and safety consequences more probable.

B. ISSUES RELATED TO CONTENTION 2:

Contention 2 states as follows:

> That the revised temperature/pressure limits that have been set for Turkey Point Unit 4 are non-conservative and will cause that reactor unit to exceed the requirements of General Design Criterion 31 of Appendix A to 10 CFR Part 50, which requires that the reactor coolant pressure boundary be designed with a sufficient margin to ensure that, when stressedd under operating, maintenance, testing, and postulated accident conditions, (1) the boundary behaves in a non-brittle manner and (2) the probability of a rapidly propagating fracture is minimized.
>
> Petitioners contend that the new pressure/temperature limits could cause the reactor vessel to exceed these requirements because the Licensee has based its calculation of the predicted RT_{NDT} for Unit 4 partly on surveillance capsule V test results from Turkey Point Unit 3 rather than prediction the RT_{NDT} for Unit 4 based on Unit 4 capsule V surveillance capsule data--a practice which is not scientific, not valid, and could cause the Unit 4 reactor to behave in a brittle manner which would make the chances of a pressure vessel failure and resultant meltdown more likely. Petitioners contend that predictions of RT_{NDT} and pressure/temperature limits derived from the shift in nil-ductility transfer should be based only on plant-specific Unit 4 data, especially in light of the fact that the only tests ever performed on Unit 4 weld specimens demonstrated that the weld material in the Unit 4 vessel was 30% more brittle than that of Unit 3. Because Unit 4's weld material is more embrittled, Petitioners contend that the FPL Integrated Surveillance program does not meet the Requirements of 10 CFR Appendix G Parts V.A and V.B, and 10 CRF Appendix H, including Appendix H Parts IIC and IIIB. Finally, Petitioners contend that the surveillance capsule V for Unit 4 should be tested to establish the new pressure/temperature limits and should the testing indicate that the RT_{NDT} for Unit 4 has passed the 300 deg.F screening criterion set by the NRC, Unit 4 should be shut down unitil it is demonstrated that the Unit 4 reactor pressure vessel can maintain its integrity beyond this limit.

The pressure/temperature limits for Turkey Point Units 3 and 4 are among the most critical limiting conditions of operation because they define the permissable operating envelope during reactor heatup, cooldown, criticality, and testing and are designed to ensure the integrity of the reactor pressure vessel, a critical piece of safety equipment.

According to 10 C.F.R. Appendix G, the pressure/temperature limits must be predicted based on the results of pertinent radiation effect studies that predict the effects of neutron irradiation on pressure vessel embrittlement. These limits are required to be based on the most limiting nil-ductility reference temperature (RTNDT) for the respective reactor units.

As explained earlier, the reference temperature is the point at which the pressure vessel metal loses nearly all of its ability to withstand shock. Thus, it is necessary to accurately and conservatively account for the effects of irradiation and other factors on the RTNDT of the pressure vessel in order to set conservative P/T limits that will protect the public from a brittle fracture of the vessel and subsequent meltdown of the reactor core.

In order to meet the requirements of Appendix G, 10 C.F.R. Appendix H requires the Licensee to establish a surveillance program to periodically withdraw surveillance capsules from the reactor vessel and test them to determine shifts in the RTNDT. This calculated shift in the fracture toughness of the vessel material due to neutron irradiation damage is called the Adjusted Reference Temperature or (ART). Appendix H also allows an integrated surveillance program for multiple reactors located at a single site on an individual case basis depending on the degree of commonality and the predicted severity of irradiation.

Contention 2 primarily contends that the current pressure/temperature limits that were set for Turkey Point Unit 4 do not meet the requirements of Appendices G and H, and that these limits are non-conservative and could cause the Turkey Point reactor Unit 4 to exceed the General Design Criterion 31 of Appendix A to 10 C.F.R. Part 50.

Intervenenors base their belief on the following issues of fact:

1. <u>The Turkey Point Unit 4 pressure/temperature limits should be set using plant specific data.</u>

Intervenors have contended throughout this proceeding that the revised Turkey Point Unit 4 P/T limits should have been based on the results of plant specific surveillance capsule test data. Intervenors base their contention on the Pacific Northwest Laboratory Report NUREG/CR-2837 entitled <u>PNL Technical Review of Pressurized Thermal Shock Issues</u>, July 1982 which states that "evaluating the failure probability of any nuclear pressure vessel is very complex. The evaluation must be plant-specific to allow for differences in material properties of the plant components, systems, configuaration, operating procedures, and dosimetry history." (Exhibit 9 at 1.1) The report also states that "predicting the material properties of plant-specific reactor vessels requires an accurate knowledge of neutron exposures of metallurgical test specimens and an accurate knowledge of the neutron exposure of plant-specific pressure vessel components." (id. at 5.11).

This view is also supported by NRC Safety Engineer, Demetrios Basdekas in a memo to Commissioners Gillinsky and Ahearne, re: Staff Report on PTS, dated December 3, 1982, wherein Basdekas states that a meaningful PTS assessment may be performed in a plant-specific basis only. (Exhibit 10 at p.3). One should note that both the analysis of P/T limits and the analysis or screening criterion for pressurized thermal shock (PTS) depend on the changes in the fracture toughness of the beltine material.

Finally, this view is further supported by Dr. George Sih, Director of Fracture Mechanics at Lehigh University who states in a letter to Intervenor's former attorney Martin H. Hodder, dated October 10, 1985, that:

> The rate at which the beltline weld material deteriorates and/or embrittles depends on the *combined* effects of irradiation and pressurized thermal shock. It is plant-specific in the sense that the influence differs inherently from one unit to another. In other words, the metallurgical properties alone cannot determine the damage behavior of the welds. The *loading history* plays a major role. Unless the rates of irradiation, fluctuation in thermal gradients and time variation in pressure are exactly the same for both Units No. 3 and No. 4, one is not justified to assume that data collected in Unit 3 could be applied to predict the behavior of Unit No. 4. Hence, conslusions drawn on change of RT_{NDT} for Unit No. 4 based on the data of Unit No. 3 cannot be considered valid. (Exhibit 11 at 2)

The need for plant specific data to be used to calculate the adjusted reference temperature (ART) to revise the Unit 4 P/T limits is especially significant in light of the fact that the only known test data concerning the actual embrittlement of Unit 4 demonstrated that the neutron damage to the pressure vessel welds in Unit 4 was far greater than anticipated and far greater than the embrittlement of the reactor vessel for Unit 3. (Exhibit 8)

Thus, Intervenors find it incredible that the NRC Staff would allow the Licensee to use data from the less severely affected Unit 3 combined with the original Unit 4 data, (which results in a smearing and diluting of the data), to predict the P/T operational limits for Unit 4.

The central issue necessary to demonstrate the basis for Intervenors' continuing concerns is the Licensee's Integrated Surveillance Program. Thus, Intervenors will address the majority of their issues of fact in their discssion of that program.

<u>2. Intervenors contend that the Licensee never met the requirements of the Integrated Surveillance Program, and they still don't meet the requirements.</u>

As explained earlier, since the fracture toughness of the reactor vessel changes as the vessel is exposed to neutron irradiation, it is necessary to periodically recalculate the P/T limits to account for changes in the fracture toughness of the reactor vessel.

This change is the fracture toughness, or adjusted reference temperature (ART) is calculated by removing surveillance capsules of weld material from the reactor units and performing charpy tests on the surveillance specimens. Appendices G and H of 10 C.F.R. require that licensee's periodically remove and test surveillance capsules to determine the shift in RTNDT.

Appendix H allows in some cases for the reactor surveillance programs to be combined and/or integrated. According to Appendix H, Section II.C there are certain criteria to be used in evaluating whether or not an integrated surveillance program is justified. The criteria are:

1. There must be substantial advantages to be gained, such as reduced power outages or reduced personnel exposure to radiation, as a direct result of not requiring surveillance capsules in all reactors in the set.

2. The design and operating features of the reactors in the set must be sufficiently similar to permit accurate comparisons of the predicted amount of radiation damage as a function of total power output.

3. There must be an adequate dosimetry program for each reactor.

4. There must be a contingency plan to assure that the surveillance program for each reactor will not be jeopardized by operation at reduced power level or by an extended outage of another reactor from which data are expected.

5. No reduction in the requirements for number of materials to be irradiated, specimen type, or number of specimens per reactor is permitted, but the amount of testing may be reduced if the initial results agree with predictions.

6. There must be adequate arrangement for data sharing between plants.

Turkey point Units 3 and 4 began operation with three capsules containing weld metal specimens in each of the Turkey Point Units -

one of capsule T, one of capsule V, and one of capsule X. Intervenors have already demonstrated that when the first weld metal capsule T specimens were tested in 1976 in order to revise the pressure/temperature limits, the tests showed that the Unit 4 weld metal was found to be the limiting material for controlling the vessel RTNDT because it exhibited a greater sensitivity to neutron radiation embrittlement in that there was about a 30% difference in calculated RTNDT. (Exhibit 6 at 27).

The SWRI report on the testing of capsule 4 also suggested that "because of the potential of reaching a low C_v shelf energy condition in the Turkey Point Unit 4 weld metal in the next few years, it is advisable to obtain another data point in the not to distant future. (Exhibit 7 at 36).

Another report by SWRI dated May 1979 and entitled Reactor Vessel Material Surveillance Program suggested that capsule V be removed from each unit after approximately 7 EFPY of operation and that the data obtained from capsule V be used to revise the P/T limits beyond 10 EFPY. (Exhibit 12).

According to the Licensee's surveillance program in existence at that time capsule V from both Unit 3 and 4 was scheduled to be removed from both units 3 and 4 and be tested on or about 1985. However, in February 1985 the Licensee requested and was later granted a license amendment which allowed them to integrate their surveillance programs for Units 3 and 4 and delayed the test of the Unit 4 capsule V surveillance specimens until 1997. (Exhibit 13).

Intervenors contend that the Licensee was improperly and perhaps illegally granted this license amendment by the Staff because the Licensee did not meet the criteria of an Integrated Surveillance Program when the amendment was granted, and they still do not meet these requirements.

First of all, the Appendix H criteria states under Section II.C(5) that the testing may be reduced if the initial results agree with the predictions. The documents presented herein prove that the test results for Unit 4 did not agree with the predictions. This view is supported by Licensee's response to Intervenors' Interrogatory B.4 where they state that the adjusted reference temperature for Unit 4 capsule T, the only tested capsule, was higher than the adjusted reference temperature predicted by Revision 1 to Regulatory Guide 1.99. Furthermore, the Affidavit of Stephen A. Collard (September 11, 1989) at 46 states that FPL had informed the NRC on several occassions prior to the Staff issuance of the Safety Evaluation on the amendments of the discrepancy in the test results for the weld capsules from Turkey Point Units 3 and 4. The Staff objected to answering Intervenors' Interrogatory No.15, which asked them why they allowed FPL to implement the Integrated Surveillance Program when results for Unit 4 capsule T did not agree with predictions.

Second, the Appendix H criteria requires that the design and operating features of the reactors in a set must be sufficiently similar to permit accurate comparisons of the predicted amount of radiation damage as a function of total power output. Intervenors contend that at the time they were permitted to implement the Integrated Surveillance Program the Licensee and the Staff realized that implementation of the flux reduction program designed to cut down on the amount of neutron irradiation bombarding the vessel walls, would mean that Turkey Point Units 3 and 4 would be operating with mixed fuel cores that were not identical in nature, and that this practice continues to date.

According to an NRC document dated February 27, 1985, re: "Near Term Flux Reduction - Turkey Point Plant Units 3 and 4" ,the flux reduction program was implemented for cycle 8 in Unit 3 and cycle 9 in Unit 4. (Exhibit 14, Enclosure 1 p.4).

A review of an FPL document entitled Reactor Cavity Neutron Measurement Program for FPL Turkey Point Unit 3, dated April 1986, states "over the lifetime of a nuclear power plant, changing fuel management schemes can result in significant changes in both magnitude and distribution of neutron flux and hence, neutron fluence throughout the reactor vessel beltline region." (Exhibit 15, pp.1-1 to 1-2).

A review of the reload Safety Evaluation documents for Unit 3, cycle 10, and Unit 4, cycle 10, demonstrate that the units were operating in cycle 9 with different fuel core mixes. For example, Unit 3 was operating in cycle 9 with 56 Westinghouse optimized fuel assemblies and 101 Westinghouse 15 X 15 low parasitic (LOPAR) fuel assemblies. (Exhibit 16). Unit 4 was operating in cycle 9 with all

Westinghouse 15 X 15 low parasitic (LOPAR) fuel assemblies. (Exhibit 17). The differences in fuel were continued in cycle 10 with Unit 3 operating with 112 Westinghouse optimized fuel assemblies and 45 Westinghouse 15 X 15 low parasitic (LOPAR) fuel assemblies, and Unit 4 with 117 Westinghouse 15 X 15 low parasitic (LOPAR) fuel assemblies and 40 Westinghouse 15 X 15 optimized fuel assemblies. (Exhibits 18 and 19).

It is interesting to note that on page 4 of the Staff's Safety Evaluation attached to the 1985 amendment granting the Integrated Surveillance Program it states that, "If future core designs are significantly different from those documented by the Licensee, the Licensee must explain the effect the changes have on neutron irradiation damage and the surveillance capsule withdrawal schedule." (Exhibit 9, p.4).

It is incomprehensible to Intervenors why the Safety Evaluation, which did not document the discrepencies in the Unit 3 and Unit 4 capsule results also does not document mixed fuel core design changes that existed at the time of issuance of the amendment and to Intervenors' best belief will continue to exist until Turkey Point Units 3 and 4 have achieved homogeneous cores some time in the future.

Intervenors contend, as does Dr. George Sih in his letter of October 10, 1985, that "loading history" plays a major role in the embrittlement process. (Exhibits 11, p.2).

Third, Intervenors also contend that the Turkey Point units have had marked differences in capacity factors in some years that could jeopardize the integrated surveillance program. Stephen Collard testifies in his affidavit at 54 that if one unit has an extended outage or period of low power operation the test data from the unit which experienced the extended outage or period of low power operation could correspond to a relatively low fluence and might not be suffcient to confirm the existing fracture toughness of the reactor vessel of the other unit. (Collard affidavit at 54).

It is interesting to note that the capacity factors for units 3 and 4 in 1984 the year before the amendment was granted were significantly different. In 1984, Unit 4 had a capacity factor of 81.8 % and Unit 3 of 52.6%. Even more striking is the fact that in 1981, Unit 4 operated at a high 78.5% capacity factor, while Unit 3 operated at a mere 16.1%. (Exhibit 20).

These divergent capacity factors continued to exist subsequent to 1985. According to Licensee's Response to Intervenors' Interrogatory B-1, in 1986 Unit 3 had a 75.9% capacity factor and Unit 4 a capacity factor of 29.7%. In 1987, Unit 3 had a capacity factor of 15.3% and Unit 4 of 45.1%.

Fourth, despite these differences in capacity factors and although Stephen Collard states in his affidavit at 55 that Appendix H to 10 C.F.R. Part 50 require each integrated surveillance program to have a contingency plan to ensure that if one unit in an integrated surveillance program has an extended outage or period of low power operation, surveillance capsule test data will be available with fluences comparable to the fluences being accumulated by the other operating units in the integrated program. Intervenors contend that the Licensee does not have an adequate contingency plan to ensure that these differences in capacity factors will not compromise the program.

Intervenors contend that this is so because in response to Intervenors' Interrogatory B.3 which asked for identification of the contingency plan, Licensee referred Intervenors to documents supplied to the NRC on February 8, 1985, and March 6, 1985 as part of their amendment request. A review of these documents suggests that FPL did not then, nor do they now, have a concrete contingency plan to meet the requirements of Appendix H. For instance in the Safety Evaluation attached to the Licensee's February 1985 letter under Contingency Plan in the Event of Reduced Power Operations or Extended Outage it states: "Both plants have capsules." (Exhibit 21 SE p.2) (Also, see Collard Affidavit at 49).

Additionally, when Intervenors reviewed documents produced by Licensee in response to Intervenors' document request, they were advised by counsel for the Licensee, John Butler, that there was no written document entitled "Contingency Plan".

Intervenors contend that the Licensee's failure to have a contingency plan to ensure that they are correctly calculating the fluence to the vessel and subsequent reduction in fracture toughness means that they do not now nor have they ever met the requirements of the integrated surveillance program.

Additionally, in relation to differences in capacity factors, Intervenors would like to address Licensee's spurious argument that even if a difference in capacity factors or EFPY were postulated to occur since 1985, and even though it would be possible for the remaining capsules in one of the Turkey Point Units to have significantly less fluence that the fluence of the reactor vessel of the other unit, such a result would only affect the ability to make predictions or extrapolations beyond 20 EFPY, since the existing surveillance data are sufficient for predictions or calculations up to 20 EFPY. (Collard affidavit at 58).

Intervenors contend that this argument is not correct, since if the fracture toughness of one unit is being compromised by the other unit, which has had a period of low operation, it would be prudent, say in the case of Unit 4 which has not been tested since 1976, to test capsule V to assure that the P/T limits are conservative. Stephen Collard himself states that schedules for removal and testing of surveillance capsules are designed to confirm the existing fracture toughness of the reactor vessel as well as to make predictions. (Collard Affidavit at 53). (Emphashis Supplied).

Additionally, Intervenors would also like to take issue with an argument that has been used for years to allow the Licensee to use an integrated program to predict radiation damage to Unit 4. That argument is the one used by Stephen Collard at 38-43 of his affidavit and by the NRC Staff on page 7 of the NRC Safety Evaluation quoting Prior Randall, which apparently attempts to justify the integrated surveillance program and discount the 1976 weld metal test results for Unit 4 by attributing the high test result to the alleged difference in flux lot number for the sample in Unit 4.

Intervenors have seen this argument used numerous times as a reason why capsule T for Unit 4 may have tested so much higher than Unit 3. In fact, this argument was first used by the Licensee in a letter to the NRC dated April 11, 1977, one year after the SWRI Unit 4 capsule T test results documented in the first part of this brief demonstrated that the weld metal Unit 4 was already highly embrittled. In their 1977 letter to the NRC, the Licensee states "However, the weldment samples for Unit 4 surveillance capsule T, although containing the same filled wire heat number, used a different welding flux lot number. Therefore, the Unit 3 capsule T sample is more representative of the Unit 4 reactor vessel." (Exhibit 21).

Intervenors have documented the fact that the Licensee used the "more representative" argument to justify using Unit 3 data for Unit 4 in response to the NRC's 10 C.F.R. 50.54 letter regarding pressurized thermal shock concerns relative to Unit 4 well before the Integrated Surveillance Program was granted. (Exhibit 22).

Yet, in response to Intervenors' Interrogatory nos. 7 and 8 the Staff responds that <u>flux lot is only of minor importance in determining the sensitivity to irradiation embrittlement.</u> (<u>Emphasis supplied.</u>)

If the Staff is correct in their statement, does this mean that the damaging test result for Turkey Point Unit 4 is really representative of the damage to the reactor vessel welds, and if it is representative does this mean that the public health and safety is being jeopardized because pressure/temperature limits have been non-conservatively set based on the less restrictive Unit 3 data ? The inconsistencies on this issue alone are simply to important for this Board to ignore. Especially in light of the fact that Unit 4 suffered from two serious overpressurization events in 1981 which could have caused undetectable flaws in the vessel making it more prone to brittle fracture when stressed. (Exhibit 23).

For all the above reasons, Intervenors contend that the Licensee does not now, nor have they ever met the requirements of the integrated surveillance program identified in Appendix H of 10 C.F.R. Part 50. Intervenors further contend that because the Licensee does not meet the requirements of the program, this Board should require them to set the pressure/temperature limits for Unit 4 based on test results from the most limiting material. The most conservative way to accomplish this would be to require the Licensee to immediately test capsule V of Unit 4 and use Unit 4 capsule T and or V surveillance specimen data to adjust the reference temperature and revise the P/T limits for Unit 4.

An alternative would be for the Licensee to calculate the ART and revise the P/T limits for Unit 4 based on only Unit 4 capsule T data but using Regulatory Guide 1.99, Revision 1. Intervenors ask this Board to reject the Licensee's argument presented at 74 of the Collard affidavit where it states that there would be little difference in the P/T limits for Unit 4 if only Unit 4 data was used. First of all, this curve was calculated roughly on a desk top computer for the purpose of the settlement discussion held between Intervenors and the Licensee. Second, neither the calculation nor the software program utilized in the detemination of the calculation have been verified by the NRC Staff or any other independent body, such as Westinghouse. Third, Collard himself states at 75 that it would be inappropriate to calculate P/T limits using only one surveillance data point for Unit 4, because such an approach would be inconsistent with Regulatory Guide 1.99. Yet, in the prior paragraph, he asks the Board to accept this exact type of hypothetical calculation as a reason for accepting the Licensee's assertion that using Unit 4 plant specific data would have little effect on the P/T limits.

Furthermore, Intervenors disagree with Mr. Collard. Intervenors contend that in the event that this Board does not agree that Unit 4 capsule V should be tested, it would be more conservative and proper to use the one Unit 4 data point and the Regulatory Guide 1.99, Revision 1 to calculate the ART and revise the P/T limits instead of Revision 2, which the Licensee used in their hypothetical calculation.

Intervenors also ask this Board to take note of the fact stated at 47 of the Collard affidavit that Turkey Point is atypical among plants with NRC accepted integrated surveillance programs in that most of the plants involved in such programs do not have surveillance capsules in their reactor vessels. Thus, one can understand the need for such a program for units that have no test capsules, but it is hard to justify such a program for Turkey Point Unit 4 which has its own test capsules, and whose initial weld tests have indicated there may be a high degree of embrittlement.

Intervenors do not believe the Licensee's argument that the integrated program will save radiation to workers meets the "there must be a substantial advantage to be gained" criterion of Appendix H. Especially in light of the fact that if all the capsules in both units are to be withdrawn over the lifetime of the units, there would be no dose savings. The dose would merely be spread out over time.

<u>3. Letter of Dr. George Sih Concerning Issues Relating to Intervenor's Contention 2.</u>

In a letter to Intervenors dated October 18, 1989, Dr. George Sih stated: "...the unit 3 data are incomplete and not sufficient to predict the P/T limits for unit 4. Additional factors such as strain rate and load-history dependent damage accumulation should be considered; they cannot be discussed on an ad hoc basis without analytical and/or experimental support."

"While the P/T limits depend on the combined effects of material properties, operating temperature and neutron irradiation as mentioned on p.7, change in *strain rate* can significantly affect the fracture toughness and RT_{NDT}. This influence has not been taken into account in determining the P/T limits."

"No confidence can be placed in determining P/T limits unless the influence of local strain rates on the fracture toughness of reactor vessel materials is accounted for or shown to be otherwise...Damage accumulation is a highly nonlinear process. Predictions based on the *linear sum* is not always conservative...In general, Turkey Point Unit 3 and 4 do differ in their load history. The information supplied by the ISP is not sufficient to conclude that the unit 3 data could be used to predict the behavior of unit 4." (Sih Letter, Attachment A).

In his letter to Intervenor, Joette Lorion, Dr. Sih takes issue with a number of Licensee's assertions. First he takes issue with Licensee's supporting argument for measuring fracture toughness described on page 7-9 in that he states that fracture toughness is strain rate dependent and cannot be adequately described by the work done in ft-lb_f. (Sih letter at 1).

Second, Dr. Sih states in a footnote on page 2 of his letter that Licensee's statement on page 14 of their motion for Summary Disposition where they state that "the rates or duration of accumulation" are not important in considering the effects of neutron irradiation appears to be in contrast with one of the most important unit nvt for measuring irradiation damage of materials. (id. at 2).

Third, on the same page, Dr. Sih states that it is not sufficient to draw conclusions from the differences in neutron fluence based on the total sum because material degradation caused by neutron irradiation being accumulative is a time history and rate dependent process. (id. at 2).

Dr. Sih further states that damage accumulation is a highly nonlinear process and thus predictions based on a linear sum are not always conservative. As evidence of this Dr. Sih uses the data supplied to Intervenors in response to Interrogatory C as a case in point. Dr. Sih points out that although the total operating time between Units 3 and 4 is only 4.8%, the deviations on a yearly base are enormous. (id. at p.3). Dr. Sih plotted these figures on a graph and showed that Unit 3 behaved very differently from Unit 4 in that it possessed a slow down period. (id. at Table 2).

Finally, Dr. Sih concluded that Turkey Point Units 3 and 4 do differ in their loading history and that the information supplied by the Integrated Surveillance Program is not sufficient to conclude that the Unit 3 data could be used to predict the behavior of Unit 4. (id. at 3).

4. Other issues for consideration by the Board

Intervenors would also like to present other issues for the Board's consideration.

The first issue concerns the fact that though both the Licensee and FPL contend that Unit 3 has more Effective Full Power Years (EFPY) than Unit 4, it is difficult to understand how this could be so in light of the fact that according to information provided in

Response to Intervenor's Interrogatory No. B.1, Turkey Point 4 has nearly 10,000 more Effective Full Power Hours (EFPH) than Unit 3 and a higher lifetime capacity factor.

Intervenors contend that the difference in capacity factor and EFPY is important because according to Stephen Collard at paragraph 51 of his affidavit, a change in EFPY or capacity factors might affect the total fluence which could affect the fracture toughness of the vessel. Intervenors would caution the Board, however, that even though these differences in capacity factor and EFPY are important, they are only some among the many factors that should be considered in determining the damage to the vessel welds. (See letter of Dr. George Sih, Exhibit 11 p.2 and Sih letter, Attachment A).

The second issue concerns the fact that the Licensee may be underestimating the calculated fluence for Turkey Point Unit 4. In a <u>Safety Evaluation Regarding Projected Values of Material Properties for Fracture Toughness for Protection Against Pressurized Thermal Shock Events</u>, attached to a letter from the NRC to FPL dated March 11, 1987, indicates that the Brookhaven National Lab (BNL) calculation for Unit 4's fluence had a 12% discrepency with Licensee's calculations as opposed to a 3% discrepency between the Licensee's and BNL's calculations of Unit 3's fluence. Thus, the Licensee could be underestimating the fluence for Unit 4. (Exhibit 24).

Additionally, Intervenors would like to suggest that if one considers the difference of fluence between capsule T from Units 3 and 4 and then associates this difference in fluence porportionally to Unit 3 capsule V, one can predict the fluence value of Unit 4 capsule V.

Fluence of Unit 4 capsule T	= 6.05×10^{18}
Fluence of Unit 3 capsule T	= 5.68×10^{18}
Capsule Fluence Difference	= 0.37×10^{18}
($3.7 \times 10^{19}/100$)	= $.037 \times 10^{19}$
Fluence of Unit 3 capsule V	= 1.229×10^{19}
Capsule Fluence Difference	= 0.370×10^{19}
Fluence of Unit 4 capsule V	= 1.599×10^{19}

Intervenors believe that the above predicted fluence of Unit 4 capsule V would produce an unacceptable P/T curve outside of conservative margins of safety embraced within the acceptable operating parameters for operation of the Turkey Point Unit 4 up to 20 EFPY and well above the $1.26 \times 10_{19}$ n/cm^2 that has been predicted for 20 EFPY. (Collard Affidavit at 57).

CONCLUSION

It is evident from the issues raised herein, that Intervenors have established that there are substantial and material issues of fact concerning Contention 2 and that these important safety issues deserve to be resolved at a public hearing.

Set in context, the available facts presented to this Board reinforce Intervenors claims that the NRC Staff and FPL have acted improperly throughout the years to avoid, rather than to confront the crucial problem of reactor pressure vessel embrittlement in

Turkey Point Unit 4 - a problem that threatens the health and safety of all who live in the south Florida area.

For all the above stated reasons, Intervenors request that this Board deny Licensee's motion for Summary Disposition of Intervenors Contention 2 and take immediate steps to investigate Intervenor's claims through a full and formal public hearing.

Intervenors would also ask that this Board revoke the subject license amendments at once because the Licensee does not meet the requirements of the Integrated Surveillance Program, the data from which served as a basis for the pressure/temperature limits established by the amendments.

Dated this 19th day of

October 1989 in Miami, Florida.

Respectfully submitted,

Joette Lorion

Joette Lorion, Director
Center for Nuclear Responsibility
7210 Red Road #217
Miami, Florida 33143
(305) 661-2165

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

'89 OCT 23 P4:33

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of	)	
	)	
FLORIDA POWER & LIGHT CO.	)	Docket Nos. 50-250 OLA 50-251 OLA
	)	
Turkey Point Plant Units 3 and 4	)	(Pressure/Temperature Amendments)

CERTIFICATE OF SERVICE

I hereby certify that copies of "Intervenors' Statement of Material Facts As to Which There Is A Genuine Issue To Be Heard" and "Intervenors' Response to Licensee's Motion for Summary Disposition of Intervenors' Contentions" with attatched letter of Dr. George Sih and Exhibits have been served on the Licensing Board by Federal Express and on the parties by deposit in the U.S. Mail, Postage Prepaid on the date shown below:

Dr. Paul Cotter
Atomic Safety & Licensing Board
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Glenn O. Bright
Atomic Safety & Licensing Board
U.S. Nuclear Regulatory Commission
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Dated: October 19, 1989

LEHIGH UNIVERSITY
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Attachment A

G. C. Sih
Director

Fax: (305) 667-3361 October 18, 1989

Ms. Joette Lorion
Center for Nuclear Responsibility
7210 Red Road, Suite 217
Miami, Florida 33143

RE: Turkey Point Nuclear Power Plant Integrated Surveillance Program (ISP).

Document A. Affidavit of Stephen A. Collard on Contentions 2 and 3 by FPL.

Document B. Licensee's Response to Intervenors' First Set of Discovery Requests to Licensee (August 7, 1989).

Dear Ms. Lorion:

Based on the package of documents you mailed me on the Turkey Point Nuclear Power Plant Integrated Surveillance Program, I find that the unit 3 data are incomplete and not sufficient to predict the P/T limits for unit 4. Additional factors such as strain rate and load-history dependent damage accumulation should be considered; they cannot be discussed on an ad hoc basis without analytical and/or experimental support.

The following comments refer to documents A and B referenced above.

(1) Pressure/Temperature Limit (Document A - Section IB7, 8 and 9 on pp. 7 to 9 inclusive).

While the P/T limits depend on the combined effects of material properties, operating temperature and neutron irradiation as mentioned on p. 7, change in strain rate $\dot{\epsilon}$ can significantly affect the fracture toughness and ΔRT_{NDT}. This influence has not been taken into account in determining the P/T limits.

The supporting argument for measuring fracture toughness from the Charpy V-notch tests is not conclusive because fracture toughness is strain rate dependent and cannot be adequately described by the work done in ft-lb_f. Work done per unit time or ft-lb_f/sec is the relevant quantity in determining damage thresholds. This is illustrated in Table 1 for the HY-80 casting material. Note that the four cases considered are the same in ft-lb_f but the applied strain rates are different. The smaller weight 30 lb_f falling through a larger distance 8 ft identified as Case IV giving rise to a higher strain rate. Comparing with Case I, a small increase in strain rate by a factor of 1.1 can lead to almost four (4) times reduction fracture toughness $(dW/dV)_c$ which is related to K_{1c} by the relation

Table 1. Influence of Strain Rate on Yield Strength and Fracture Toughness Determined from Three-Point Bent Specimen as Specified by ASTM E-23 for HY-80 Casting Material. (Ref. G. C. Sih and D. Y. Tzou, "Dynamic Fracture Rate of Charpy V-Notch Specimen", Journal of Theoretical and Applied Fracture Mechanics, Vol. 5, pp. 189-203, 1986).

Case No. ($ft\text{-}lb_f$)	Strain Rate $\dot{\epsilon}$ (sec^{-1})	Yield Strength σ_{ys} (ksi)	Critical Energy Density $(dW/dV)_c$ (ksi)
I (1 x 240)	70.36	78.28	24.46
II (2 x 120)	74.00	79.15	15.70
III (4 x 60)	74.80	80.02	10.08
IV (8 x 30)	77.36	80.90	6.47

$$\left(\frac{dW}{dV}\right)_c = \frac{(1+\nu)(1-2\nu)K_{1c}^2}{2\pi r_c E}$$

where ν and E are, respectively, the Poisson's ratio and Young's modulus. The last ligament of material that triggers fast fracture is r_c.

The local strain rates in the reactor vessel where defects prevail can be high and cannot be known unless a two-dimensional, if not three-dimensional, non-linear elastic-plastic stress analysis is performed. No confidence can be placed in determining P/T limits unless the influence of local strain rates on the fracture toughness of reactor vessel materials is accounted for or shown to be otherwise. This effect cannot be dismissed on an ad hoc basis because it affects the calculations of ART, ΔRT_{NDT}, etc.

(2) Neutron Irradiation (Document A - Section IIIB 61 to 65 inclusive on pp. 42 to 44).

Referring to the data on neutron fluence (n/cm^2) in Table 5 on p. 43, it is not sufficient to draw conclusions from the difference of 3.6×10^{17} n/cm^2 (life time) and 2.37×10^{17} n/cm^2 (1965-90) between unit 3 and 4 based on the total sum. Material degradation caused by neutron irradiation being accumulative is a time-history and rate dependent process. It would be more informative to investigate the rate* at which the neutron fluence is accumulated in time on monthly or at

*The materials on p. 14 of Licensee's Motion for Summary Disposition of Intervenors' Contentions state that ---"the rates or duration of accumulation---" are not important in considering the effects of neutron irradiation. This statement appears to be in contrast with one of the most important unit nvt for measuring irradiation damage of material. Here, n stands for the number of neutrons per cm^3, v the velocity in cm/sec and t the time. Rate effect is reflected by v and and duration by t.

least yearly basis. This point will be highlighted in relation to EFPH.

Damage accumulation is a highly nonlinear process. Predictions based on the *linear sum* is not always conservative. The data in Table 5 are not supportive of the integrated surveillance program.

(3) Annual EFPH (Document B - Section on Licensee's Response C on p. 11).

A case in point on the influence of rate effect can be illustrated by the annual EFPH data on p. 11 which is summarized in Table 2. Although the differ-

Table 2. Annual EFPH for Turkey Point Unit 3 and 4 from 1985-88.

Year	Unit 3	Unit 4	% Deviations
1985	5,032.5	7,706.5	+ 53.1
1986	6,652.9	2,601.8	- 60.9
1987	1,344.6	3,950.2	+193.8
1988	5,176.3	4,828.9	- 6.7
	18,206.3	19,087.4	+ 4.8

ence in the total operating time between unit 3 and 4 is only +4.8%, the deviations on a yearly basis are enormous. A graphical representation of the data in Table 2 can be found in Figure 1. Unit 3 behaved very differently from unit 4; it possessed a slow down period. The two curves intersected at P between 1986 and 1987 aside from the initial crossing. An overestimate would result to the left of P and underestimate to the right of P should the data of unit 3 be applied to predict that of unit 4. The net damage would not add and subtract as in arithmetic.

In general, Turkey Point Unit 3 and 4 do differ in their load history. The information supplied by the ISP is not sufficient to conclude that the unit 3 data could be used to predict the behavior of unit 4.

Very sincerely yours,

George C. Sih
Professor of Mechanics

GCS:bd

Enclosure: Figure 1

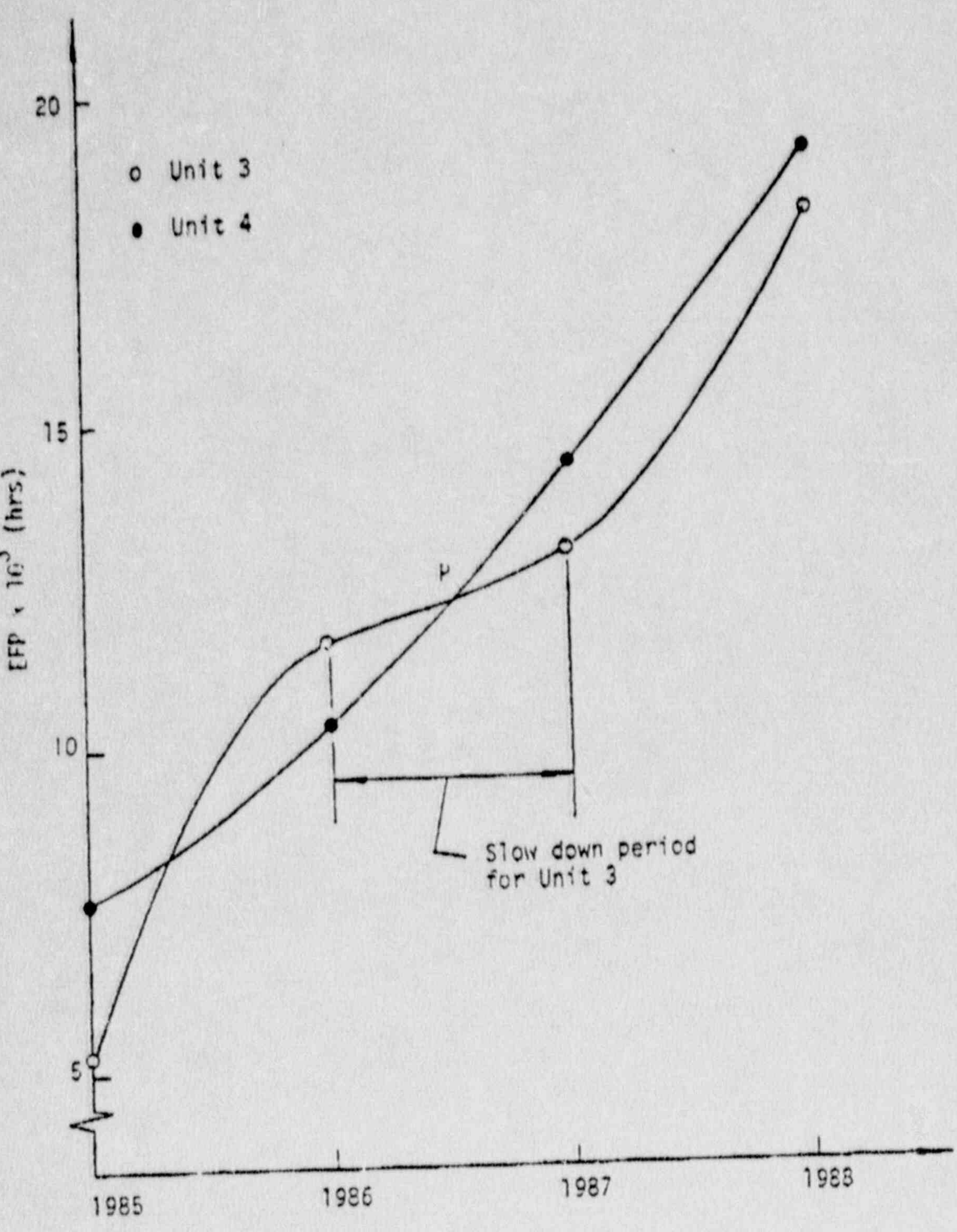


Figure 1. Data reproduced from section (c) on page 11 of Licensee's Response to Intervenors' First Set of Discovery Requests to Licensee (August 7, 1989): Docket Nos. 50-250 OLA-4 and 50-251 OLA-4.

Biography

of

Dr. George C. M. Sih

Professor of Mechanics and Director of the
Institute of Fracture and Solid Mechanics

Dr. Sih is currently Professor of Mechanics and Director of the Institute of Fracture and Solid Mechanics at Lehigh University, Bethlehem, Pennsylvania. He also holds the appointment of Adjunct Professor at The Hahnemann Medical College and Hospital of Philadelphia since 1972. He received his B.S. at the University of Portland, Oregon, 1953; his M.S. at New York University, 1957; and Ph.D. at Lehigh University, 1960; all of these degrees in Mechanical Engineering.

Dr. Sih has engaged in research in the interaction of mechanical deformation and heat flow (1960) supported by the Koppers Foundation, in Fracture Mechanics (1960 and 1961) for the Boeing Company Transport Division and (1962 to 1965) for the National Science Foundation, and as a member of the Technical Staff, Bell Telephone Laboratory (Summer 1961). He has been engaged as Principal Investigator in more than fifty projects at Lehigh University sponsored by the Office of Naval Research, Naval Research Laboratory, the National Aeronautics and Space Administration, the Air Force, the Army, etc., all of which are concerned with optimizing the use of high performance material with design, a discipline that has been frequently referred to as "Fracture Mechanics". Much of his work has been concerned with estimating the remaining life of material and structural components damaged by yielding and/or fracture. He specializes in developing computer software for predicting the mechanical behavior of structures and the stability of objects moving through fluid media. His more recent activities are concerned with the influence of moisture and temperature in composite materials, laser glazing techniques and non-destructive testing methods involving high-voltage electrophotography.

From 1953 to 1957, Dr. Sih was employed by Radio Corporation of America as a project and research engineer. He worked on the research and development of input and output devices for the first generation "Bizmark" computer system. Among the significant patents he obtained were:

1. Adjustable optical system for line printing.
2. Automatic magnetic disc printing device for the Xerox process.

In 1957 and 1958, Dr. Sih returned to the academic life and served at the City College of New York as Lecturer in Mechanical Engineering. He came to Lehigh University in 1958 as Instructor in Engineering Mechanics and was appointed Assistant Professor after completion of his doctorate. From 1965 to 1966, Dr. Sih held the position of Visiting Professor in Aeronautics at the California Institute of Technology and participated in an Air Force research project on the dynamics of crack propagation and size effects in the fracture of plates.

Dr. Sih assumed in 1970 the duties of Regional Editor, International Journal of Fracture Mechanics, and the responsibilities of soliciting and reviewing papers in the field of Fracture Mechanics. From 1971 to 1975, he served as an Associate Editor of the ASME Journal of Applied Mechanics. He is also on the Editorial Advisory Board of the Journal of Engineering Fracture Mechanics. He is also Editor-in-Chief of an International Journal of Theoretical and Applied Fracture Mechanics. Dr. Sih is a Fellow of the American Society of Mechanical Engineers and Honorary Fellow of the International Congress of Fracture. He is also a founding member of the International Cooperative Fracture Institute, an organization established to promote the interchange of ideas and information among active researchers in fracture mechanics.

Dr. Sih is also a member of the following societies:

1. Society of Sigma Xi
2. ASTM Committee E-24 on Fracture Testing of Materials
3. International Society of Engineering Science
4. American Society of Civil Engineering
5. American Society of Mechanical Engineering
6. International Society for the Interaction of Mechanics and Mathematics

Dr. Sih is the Editor of three book series. Seven volumes on the Mechanics of Fracture series have been or are about to be published:

Volume I - Methods of Analysis and Solutions to Crack Problems, 1973

Volume II - Three-Dimensional Crack Problems, 1974

Volume III - Plates and Shells with Cracks, 1976

Volume IV - Elastodynamic Crack Problems, 1976

Volume V - Stress Analysis of Notch Problems, 1976

Volume VI - Cracks in Composite Materials, 1980

Volume VII - Experimental Evaluation of Stress Concentration and Intensity Factors, 1980

The two other series are Fatigue and Fracture:

Volume I - Fatigue and Fracture, S. Kocanda, 1978

Volume II - Fracture Micromechanics of Polymer Materials, V. S. Kukshenko and V. P. Tamuzh, 1980

and Engineering Application of Fracture Mechanics:

Volume I - Fracture Mechanics Methodology: Evaluation of Structural Components Integrity, edited by G. C. Sih and L. Faria

Volume II - Mixed Mode Crack Extension by E. E. Gdoutos

Volume III - Fracture Mechanics of Concrete: Material Characterization and Testing, edited by A. Carpinteri and A. Ingraffea

Volume IV - Fracture Mechanics of Concrete: Numerical Analysis and Structural Application by G. C. Sih and A. DiTommaso

Volume V - Bonded Repair of Aircraft Structure by A. A. Baker and R. Jones

Volume VI - Crack Growth and Material Damage in Concrete: Limit Load and Brittle Fracture by A. Carpinteri

Dr. Sih has also served as principal organizer and editor of proceedings of several conferences:

1. International Conference on "Dynamic Crack Propagation", (1972), Lehigh University
2. International Conference on "Prospects of Fracture Mechanics", (1974), The Netherlands
3. Conference on "Linear Fracture Mechanics", (1975), Lehigh University
4. International Conference on "Fracture Mechanics and Technology", (1976), Hong Kong
5. 14th Annual Meeting of the Society of Engineering Science, (1977), Lehigh University
6. First USA-USSR Symposium on "Fracture of Composite Materials", (1978), USSR
7. International Conference on "Fracture Mechanics in Engineering Applications", (1979), India
8. International Conference on "Analytical and Experimental Fracture Mechanics", (1980), Italy
9. International Conference on "Defects and Fracture", (1980), Poland

10. International Conference on "Mixed Mode Crack Propagation", (1980), Greece

11. International Conference on "Absorbed Energy and/or Specific Strain Energy Density Criterion", (1980), Hungary

12. International Conference on "Defects, Fracture and Fatigue", (1982), Canada

13. International Conference on "Fracture Mechanics Technology Applied to Material Evaluation and Structure Design", (1982), Australia

14. International Conference on "Application of Fracture Mechanics to Materials and Structures", (1983), Germany

Dr. Sih has approximately two hundred publications principally in the area of solid and fracture mechanics. He has authored and co-authored a total of three books.

1. Handbook of Stress Intensity Factors, 1973
2. Three Dimensional Crack Problems (with M. K. Kassir), 1974
3. Cracks in Composite Materials (with E. P. Chen), 1980

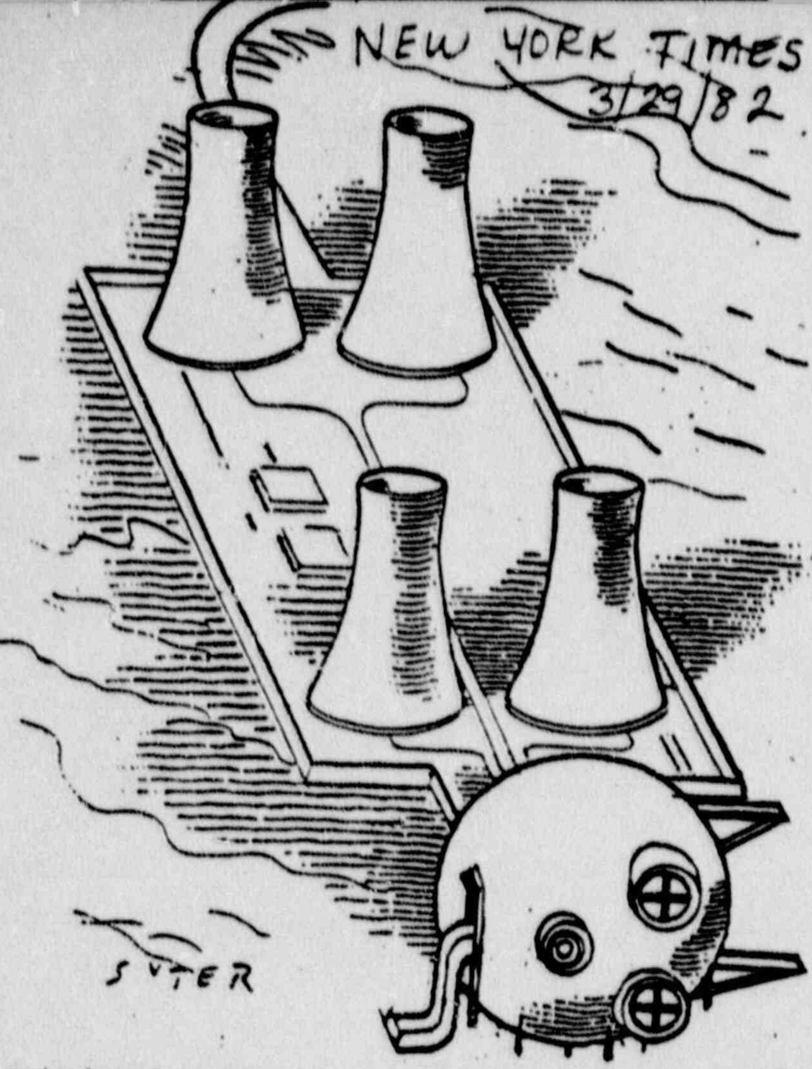
Dr. Sih received the 1975 Achievement Award from the Chinese Institute of Engineers in the United States and the 1984 Achievement Award from the Chinese Engineers and Scientists Association of Southern California for his accomplishments in research and teaching in fracture and solid mechanics.

Dr. Sih has also been active in serving as members of national committees. Among them are the National Materials Advisory Board concerning with the Dynamic Response of Materials Subjected to High Strain Rate Loading; Ship Materials Fabrication and Inspection; and other committees concerning Nuclear Reactor Components.

NEW YORK TIMES 3/29/82

The Risk Of a Meltdown

By Demetrios L. Basdekas



WASHINGTON — There is a high, increasing likelihood that someday soon, during a seemingly minor malfunction at any of a dozen or more nuclear plants around the United States, the steel vessel that houses the radioactive core is going to crack like a piece of glass. The result will be a core meltdown, the most serious kind of accident, which will injure many people, destroy the plant, and probably destroy the nuclear industry with it.

On the third anniversary of the Three Mile Island accident, the Government and industry are unable or unwilling to deal honestly and urgently with far-reaching nuclear-safety problems.

Another serious accident is very likely because the wrong metal was used in the reactor vessels, and with each day of operation, neutron radiation is making the metal more brittle, and more prone to crack in case of sudden temperature change under pressure.

One manufacturer of nuclear reactors has reported to the Nuclear Regulatory Commission that in three to five more years, the vessels in some plants will be too brittle to operate safely. But this estimate is wishful thinking, based on unrealistic assumptions about plant operators' actions and accident sequences. Some plants are already too dangerous to operate without corrective measures.

The commission could do a great deal to prevent such an accident, and stretch out the lives of many of these brittle vessels, if it ordered the type of corrective steps already taken at some European reactors. But the commission, regulating an industry that has serious financial and technical problems, instead of taking initiatives tends to sweep difficult technical problems under the rug, reacting to crises only after they occur.

The commission must realize that this crisis is upon us. A temperature change severe enough to crack a brittle reactor vessel already has occurred, in California, but not at one of the older, more vulnerable plants. The commercial nuclear industry's admirable safety record — no deaths caused by radiation — still is intact, but this cannot last much longer, because the reactor vessels and other critical components are aging.

For many years, it has been known that vessels are becoming brittle. What makes the problem urgent is that the metal is aging more rapidly than expected, and the circumstances that would cause such an accident now seem more likely.

At the Rancho Seco plant, near Sacramento, Calif., in March 1978 a worker dropped a small light bulb into an instrument panel, causing an electrical short circuit. The short wreaked havoc on the plant's control systems — a variety of instruments that run crucial pumps and valves — and the result was that too much water was pumped through the reactor, chilling it suddenly. It is very doubtful that some of the older plants operating today would be able to withstand the same shock. Fortunately, Rancho Seco had been in operation less than two years; had it been in operation for 10, its pressure vessel most likely would have ruptured.

The kinds of control systems that went haywire at Rancho Seco are very likely to fail at crucial times in other nuclear-power plants. When a pipe bursts, or a seal fails, or a valve sticks, automatic control and safety systems almost instantly take action to compensate, but they do not always take the right action.

Control systems are not reviewed by the Nuclear Regulatory Commission. They are not immune to fire or power failure; they often have no backups, so are prone to simple failure. They are not even earthquake-proof.

The N.R.C. staff has taken the position that if a plant gets into trouble because of control-system malfunctions, it has safety systems to take care of any problems. But this is not so, as events of the last few years show. At Rancho Seco, at Three Mile Island, and at other plants, control systems not thought vital to the safe operation of a plant ended up causing serious problems.

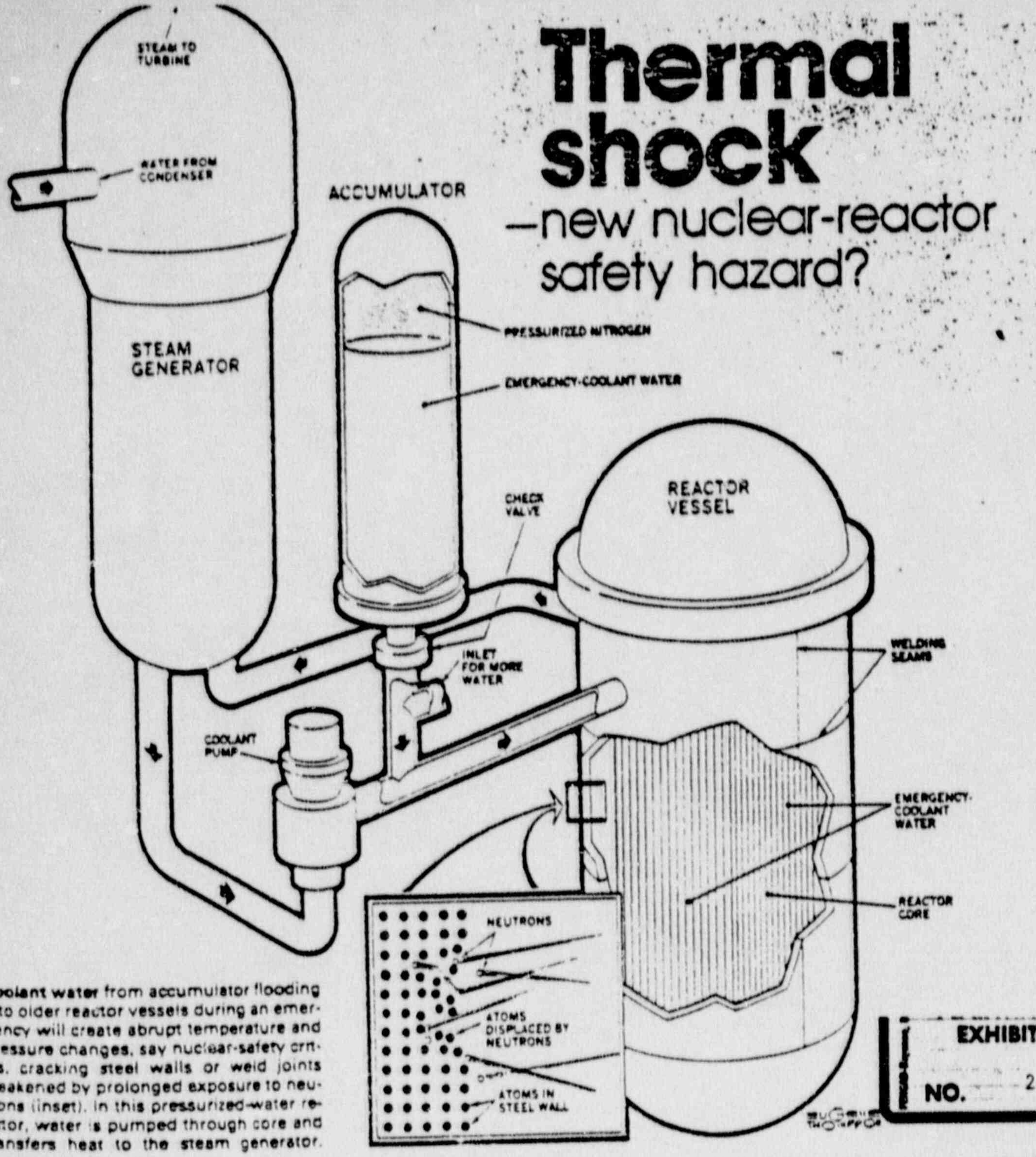
The Nuclear Regulatory Commission is charged with ensuring that nuclear plants are operated "with adequate protection" of the public health and safety. But bureaucratic foot-dragging and preoccupation with public relations and financial problems of the industry are contributing to a shortsighted view — that technical problems can wait or do not exist. Some members of the staff acknowledge the safety problems associated with control systems, but the agency has yet to demand from utilities operating nuclear-power plants the technical data on control systems necessary to assess the systems' safety fully.

It may be that we need nuclear power to maintain our standard of living. But there is a vast difference between having to accept something, and making it acceptable. We can make nuclear power acceptable.

The Nuclear Regulatory Commission chairman, Nunzio Palladino, has spoken of cleaning up our nuclear act. As a private citizen, I hope that we do so, beginning with vigilance at the N.R.C. One more accident the size of Three Mile Island's, and the public's reaction almost certainly will foreclose the nuclear option.

Demetrios L. Basdekas is a reactor safety engineer with the Nuclear Regulatory Commission.

Thermal shock —new nuclear-reactor safety hazard?



Coolant water from accumulator flooding into older reactor vessels during an emergency will create abrupt temperature and pressure changes, say nuclear-safety critics, cracking steel walls or weld joints weakened by prolonged exposure to neutrons (inset). In this pressurized-water reactor, water is pumped through core and transfers heat to the steam generator.

EXHIBIT
NO. 2

Could cooling water rupture brittle reactor walls? Here are the facts

By EDWARD EDELSON
DRAWING BY EUGENE THOMPSON

"There is a high, increasing likelihood that someday soon, during a seemingly minor malfunction at any of a dozen or more nuclear plants around the United States, the steel vessel that houses the radioactive core is going to crack like a piece of glass. The result will be a core meltdown, the most serious kind of accident, which will injure many people, destroy the plant, and probably destroy the nuclear industry with it."—Demetrios L. Basdekas, *The New York Times*, March 29, 1982

Basdekas, a reactor-safety engineer with the Nuclear Regulatory Commission, continued his article to warn that radiation is making the metal reactor vessels at some nuclear plants brittle. As a result, he wrote, water used to flood and cool reactor cores in an emergency could cause a meltdown instead of preventing one. The cause: abrupt changes in reactor pressure and temperature—a condition called pressurized thermal shock—would crack brittle vessels, allowing emergency water to escape.

The safety engineer's "piece-of-glass" charge quickly focused attention on thermal shock:

- The NRC commissioners held a public meeting.
- Rep. Ed Markey of Massachusetts called a congressional hearing

Continued

• Work on what was supposed to be a definitive study of the thermal-shock issue was accelerated by the NRC.

And the kind of debate that has become quite familiar in recent years has predictably erupted. Electrical utilities, reactor manufacturers, and the Nuclear Regulatory Commission say that the pressurized-thermal-shock problem is well in hand and that the "piece-of-glass" charge is absurd. Critics say that the nuclear people are talking through their hats because there simply isn't enough information available to assess the danger of pressurized thermal shock.

I've recently talked to experts on both sides of the question. At the moment there are no pat answers. But information about the hazard of thermal shock is accumulating steadily. Here is what you need to know.

Pressurized thermal shock has been widely publicized only recently. But inklings of a problem emerged in the 1960s.

At one power-plant reactor, a worker peered into a video monitor and manipulated a robotic arm down into the radioactive water of a 40-foot-high reactor vessel. He slowly fished out a small basket hanging near the thick metal wall of the reactor. Inside the basket was a jumble of pencil-size steel bars, each alloyed with various metals and each bearing a V-shaped notch.

At a nearby test area, he carefully unloaded his irradiated catch behind shielded-glass windows. Deft maneuvers with another robotic arm positioned each steel bar under a wedge-shaped hammer. Then, as samples were cooled or heated, he pushed a button, and the hammer slammed into the notches.

This routine Charpy test (named for its developer) yielded expected results: At lower temperatures, where metals become brittle, samples broke easily. Higher temperatures—like those in your kitchen oven—made the steel more ductile. Heated steel samples absorbed more hammer energy before snapping.

But something unexpected occurred when the worker slammed his test hammer onto bars alloyed with tiny amounts of copper. The steel—even warmed—broke easily. He raised the temperature. Still the brittle bars snapped. Finally at about 300 degrees F, the bars became ductile instead of brittle. The presence of copper seemed to be producing strange results. Soon workers at other power and research reactors discovered the same unexpected embrittlement.

What puzzled everyone was the speedup of embrittlement because of the presence of copper, not the results of the standard Charpy tests on exposed metal samples. This technique—gradually changing metal temperatures and measuring how much hammer energy the metal can absorb without breaking—actually tests radiation damage. Radiation tends to make all metals brittle; irradiated metal must be raised to a higher temperature before it will become ductile. This shift in the transition temperature from brittle to ductile is a measure of radiation damage.

Nuclear researchers, aware of metal embrittlement, had earlier exposed samples to intense radiation. But the surge of reactor construction beginning in the 1960s found engineers without enough reliable data. To answer questions about long-term radiation effects on metal, baskets of Charpy samples had been positioned in early reactors.

"Copper was used to prevent rust. Someone probably got a prize for the suggestion"

The principal cause of embrittlement was known to be neutrons, the atomic particles emitted by nuclear fission in the reactor core, colliding with metal in the reactor. "It's like billiards," says one expert. "Although metal atoms are much heavier than neutrons, when a high-energy neutron collides with a metal atom, the neutron forces the atom from its lattice—the geometric array of atoms."

The Charpy tests of the 1960s revealed that just a little copper in a steel alloy hastens embrittlement. Since that time, though, researchers have been uncertain why the presence of copper hastens radiation damage. Theodore U. Marston, who works on thermal shock at the Electric Power Research Institute in Palo Alto, Calif., says there's now strong evidence that neutron bombardment makes the copper clump together.

"Copper starts out in a solid as atoms fairly evenly distributed. Under radiation the atoms tend to come together as copper particles," he said. New instruments that let researchers see atoms within metals show this clumping effect, Marston says.

As the first discoveries of brittle irradiated steel containing copper became known, anxiety began to spread. How much copper was in the steel-alloy walls of reactor vessels across the country? Reactor-vessel manufacturers and utilities began leafing through old files to find what information they had about the copper content of metals in reactors.

Records showed that there was some copper in the vessel walls themselves. "We used a lot of auto stock," explained Marston. "When you melt it, you can't get all the wiring out."

But welds in vessel walls were the real problem. Before the industry realized what was happening, which was about 1972, spools of copper-coated welding wire were routinely used for these welds. "The copper was used to prevent rust," noted Stephen H. Hanauer, director of safety technology at the NRC. "Someone probably got a $10 prize for the suggestion."

Reactor builders switched to nickel-coated electrodes, but they couldn't replace the welds in older reactors. When I visited Marston last winter, the significance of those welds became clear. On his desk was a slab of metal that looked like a paperweight gone wild. I thought it was eight inches wide. But it was really eight inches thick—the thickness of a reactor-vessel wall. The weld was a yellowish stripe in the steel, tapering from three inches thick on one side to two inches on the other. Marston told me that it can take three weeks of repeated passes with electrodes to complete one of those welds. That type of weld, engineered to be a powerful bond between huge steel sections of reactor vessels, contained enough copper to become a potential hazard instead.

Interest in reactor-vessel embrittlement heated up in 1977, Marston recalls. There was trouble with the sample holders in a reactor built by Babcock and Wilcox, one of the major suppliers, he says. Vibration kept knocking them loose. All the samples were taken out, and "it looked worse than we thought," Marston said, indicating that embrittlement was progressing faster than expected in the test samples.

Added to this continued confirmation of embrittled-metal samples and copper contamination of vessels was an event the following year that, for some, increased the alarm.

On March 20, 1978, a worker at the Rancho Seco nuclear generating plant near Sacramento, Calif., dropped a light bulb into an instrument panel. The panel shorted out and the plant's instruments went haywire, flashing fake signals to the control systems. Rancho Seco's emergency cooling system kicked into operation. Cold water

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flooded into the reactor, dropping the temperature from 582 degrees F to 335 in a little more than an hour.

Pressure inside the reactor vessel first dropped from the normal 2,200 pounds per square inch to under 1,600 psi. Then, as high-pressure water pumps were triggered, the pressure went back over 2,000 psi. With no reliable instrumentation to guide them, control-room technicians kept the cold water flowing, maintaining the combination of unexpectedly low temperature and high pressure for several hours.

The Rancho Seco "transient," as nuclear engineers call it, made it clear that pressurized-water reactors were susceptible to abrupt changes in temperature and pressure. Could any pressurized reactors already have small cracks? And could vessel walls containing such cracks, subjected to sudden changes of temperature and pressure during an accident, then rupture, draining the coolant water and producing a catastrophic meltdown of the core?

The truth is that nobody knows for certain. Calculations indicate that under pressurized-thermal-shock conditions, a reactor vessel will fail only if cracks of a certain dimension are present on the inside wall. Inspections throughout the industry have used ultrasound and other nondestructive testing methods and thus far have found no such cracks. Industry representatives say they are reasonably confident that no cracks are there. Critics say the inspection equipment isn't good enough to detect the cracks. The NRC says its analyses assume that some cracks exist, no matter what inspections show.

Richard Cheverton of the Oak Ridge National Laboratory, whose team has performed many of the thermal-shock analyses, says assumptions about weaknesses in nuclear power plants had to be made. Take the critical issue of cracks in the reactor-vessel walls. "It's difficult to look for flaws after the reactor is in operation, and it's still a question of how good a job one can do," Cheverton said. "It's not clear yet whether some of the shallow flaws that can get us into trouble can be found with accuracy, so we tend to assume that the flaws will be there."

But Richard J. Sero, who heads a program on thermal shock for Westinghouse (a major plant builder), maintains that there is growing evidence to support the belief that the cracks aren't there. Engineers often inspect working-reactor vessels with ultrasound equipment, whose echoes are analyzed to detect anything unusual in the vessel wall—a crack, an inclusion of different material in the metal, an unevenness in the surface.

Ultrasound inspection is complicated somewhat by the fact that reactor vessels have a ⅛-inch-thick cladding—a permanently bonded layer—of stainless steel on the inside surface that can produce false echo patterns. But that's not an insuperable problem. Sero says he's impressed by the sensitivity of the equipment.

"We've done about a half-dozen full-vessel inspections," Sero said. "You do pick up what we call 'indications'—as many as 20 in some vessels. When you pick up any anomalies at all, you must look at your pre-service inspection to see if they existed before and what size they were.

"We've found that the equipment can pick up things like layers in the cladding," Sero continued. "When we've gone to the inspection reports, we've found that there are layers in the cladding at the same depth of the indication. Our conclusion is that in all the inspections we've done, we haven't found any indications that we can't resolve as inclusions of different material or layers."

"The NRC may consult its Ouija board and get a number, but the error bands are so large, it's useless"

Sero says Westinghouse gained confidence in the inspection results when one test showed a gouge on the outside wall of a reactor vessel. "We were able to get pictures of the reactor vessel that were taken before it was installed," he said. "We found that it was a gouge that existed before it went to the plant." A sample of a vessel wall containing a crack is used to calibrate instruments.

The NRC recently released a detailed study on pressurized thermal shock and reactor safety. If you really want a good fight, ask people about the reliability of those safety estimates. The method the NRC and the industry uses is called probabilistic risk assessment. It's designed to get around a rather impressive lack of concrete evidence. All the calculations about pressurized thermal shock, for example, are based on just eight events that have occurred at nuclear plants, including the Rancho Seco transient and the most famous incident of all, Three Mile Island.

In a probabilistic risk assessment, you estimate the likelihood of an event that initiates a transient, then estimate the likelihood of the reaction to that event, the reaction to that reaction, and so on down the line.

Westinghouse, for example, has a computer analysis that starts with 17 possible initiators and runs through event trees to more than 8,200 end points. The NRC has done the same thing. Its numbers come out more or less in agreement about the risk of thermal shock. But there are inevitable differences of opinion about the value of those calculations, which show that although there is no clear and present danger, corrective action should be taken at some reactors to reduce the hazard of thermal shock.

Not everyone agrees with the calculations. "The NRC may consult its Ouija board and come up with a number," said Robert Pollard of the Union of Concerned Scientists, "but the error bands on it are so large that it's essentially useless."

That's not exactly so, says Cheverton of Oak Ridge. "It's possible to estimate what the uncertainty in the analysis is, and you have to live with that uncertainty," he said. "But you take the conservative end of it and work with that."

A lack of data is more or less conceded all through the NRC report. "Perhaps the most significant uncertainty in the treatment . . . is that there are known low-frequency potential over-cooling events much more severe than those that have occurred," the report says at one point. "Because these events have not occurred, they have not been taken into account in the frequency distribution." In other words, it's tough to predict the possibility of something that has never happened. In another section, the report notes "substantial uncertainties" in some estimates and calculations that are uncertain by "plus or minus at least two orders of magnitude, a broad band of uncertainty, indeed."

What else can we do? the NRC people ask. "It isn't well defined, but it's the best information we have," said the NRC's Hanauer.

Your best is none too good, the critics say. They point out that the probabilistic-risk-assessment technique is the same one used in the famous Rasmussen report of 1974, in which a team headed by MIT professor Norman Rasmussen calculated the risks of nuclear accidents. Rasmussen came up with some comfortingly low-risk figures. Just last year, though, the

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NRC looked over the operating data that have accumulated since then and concluded that the odds of a nuclear accident occurring calculated by Rasmussen were low by a factor of 30.

Hanauer says that risk calculators have learned a lot from Rasmussen's pioneering effort. "He kissed off earthquakes in two pages and floods in two lines," Hanauer noted. Taking one volume of a shelf-long safety assessment of the Indian Point reactor near New York City, Hanauer pointed out that earthquakes and floods were toward the top of the list of risks. The NRC has learned to include such risks in its risk assessments, Hanauer says.

But Basdekas dismisses the report as "the quantification of wishful thinking." And George Sih, director of the Institute of Fracture and Solid Mechanics at Lehigh University, says that the impressive report is built on a foundation of sand.

"The samples they study are five inches long, and the vessels are 500 inches long," Sih said. "The sample is very thin, and the vessel is eight inches thick. We don't know how to transfer small-sample data to the design of large-scale structural components. The scaling effect in size and also the scaling effect in time are among the most difficult questions we have."

If critics think the NRC has been too speculative, industry believes the report is too conservative. You can arrive at just about any conclusion you want by putting in the appropriate numbers, Marston says. "By changing the assumptions," he explained, "I can show that one of these things has no useful life at all or a lifetime of 30 to 40 years." The NRC consistently takes the most conservative numbers for its estimates, he says.

One of the key factors that the NRC's experts looked at was the transition temperature at which a piece of metal stops being ductile and becomes brittle enough to break easily. A crucial part of the NRC report was to set a point at which this transition temperature in a given reactor would be cause for concern. The report sets the danger point at 300 degrees F for vertical welds, 270 degrees for horizontal ones.

Higher transition temperatures are worse, since the reactor vessel must be maintained at these temperatures if the effects of brittle metal are to be avoided. The original standard for nuclear reactors was no more than 200 degrees F. The temperature is higher for vertical welds because pressure tends to force the welds out, increasing the possibility that a crack will break through the vessel wall.

Determining a transition temperature depends on the composition of a metal, the amount of radiation it receives, and, most controversially, the stresses to which it is exposed. The NRC staff used a formula to predict how assumed pre-existing cracks might extend into the vessel wall.

As a result of tests on the rate of embrittlement at various plants, the NRC predicted when some of them will reach a danger point. All things considered, the NRC report reached a reasonably comforting conclusion. It listed 40 pressurized-water reactors in which pressurized thermal shock was an issue. "If no one does anything, we've got one reactor that's in big trouble, four others that are a little behind it, and four that are in a mild kind of trouble," Hanauer told me. "The rest of them will not reach the screening criterion [the transition temperature] during the anticipated life of the plant."

> "Though the inner portion is brittle, the outer portion is tough; radiation damage in the wall is attenuated"

The "big-trouble" generating plant is the H. B. Robinson 2 reactor of Carolina Power and Light. Hanauer calculated that if nothing were done, it would reach the transition-temperature criterion in September of 1987. Turkey Point 3 and 4 in Florida get there in 1988; Calvert Cliffs 1 in Maryland gets there in 1989; and Fort Calhoun in Nebraska arrives in 1990. Rancho Seco, Maine Yankee, Oconee 2 in South Carolina, and Three Mile Island 1 arrive in the 1990s. Everything else is 21st century, Hanauer says.

Reactor manufacturers accepted those numbers without too much argument. "Their conclusions are more or less in line with ours," said Sero of Westinghouse. Sero says that Westinghouse thinks the NRC could set its transition-temperature numbers about 30 degrees lower, but he isn't arguing with the basic premises of the report.

Nuclear critics are. They center their fire on the vast number of assumptions that had to be made in the report because information about the probability of different events occurring and about the reliability of safety systems simply isn't available. Rep[...] that the risk-assessment technique was "like predicting the winner of the World Series after the first exhibition game."

There's also a lot that the utilities and manufacturers can do to lessen any possible danger, industry experts say. One easy step is to reshuffle the fuel elements in the reactor core, putting older fuel elements, which emit fewer neutrons, close to the vessel wall. "It's easy and cheap to reduce neutron flux by a factor of two," acknowledged Hanauer.

Critics say that repositioning the fuel elements isn't enough. They want American utilities to reduce neutron exposure even further by inserting dummy fuel elements next to the vessel wall. That's been done at two reactors in West Germany and one Russian-built reactor in Finland. But utilities are reluctant to take the reduction in generating capacity that dummy fuel elements bring.

There are many other steps that can be taken, Marston said. One is the marvelously simple measure of heating the emergency cooling water to reduce thermal shock. Keeping the emergency water supply at 120 degrees F rather than room temperature is cheap and effective, Marston says. Thermal shock can also be reduced by adding controls to throttle back the automatic-feedwater system, he notes.

Improved training for reactor operators is another industry option. The idea is to get them ready for all the problems that could lead to a significant transient, then avoid the sequences that end in serious trouble.

The last resort is annealing. The reactor would be shut down, all the fuel elements would be removed, and the vessel would be heated to 850 degrees F for a week. A study done by Westinghouse for the Electric Power Research Institute concluded that annealing would make the vessel walls young again. The process isn't cheap. One report cited costs of $60 million or more for a single reactor, including the price of the electricity that the plant did not generate during the treatment.

No one is thinking about annealing right now. Instead, utilities and manufacturers are making detailed studies of all the factors affecting the thermal-shock issue for individual plants. The NRC report has asked for such a plant-specific report at least three years before a reactor reaches its screening criterion for danger.

For the Robinson 2 reactor, the report would be due in 1984. Carolina Power and Light is hard at work, says

Thomas S. Elleman, who is in charge of nuclear safety. The vessel wall has been inspected, and no cracks were found. New training for reactor personnel is under way. The company is studying a proposal to heat the emergency water supply.

Neutron exposure has been reduced by putting the older fuel elements next to the reactor wall. How much extra time will the program buy? "It's premature to speculate about that," Elleman said.

There's no panic at the NRC, the manufacturers, or the utilities. The problem is well understood, Cheverton says, and the Oak Ridge analysis indicated that even if worse came to worst, a reactor vessel would not break wide open. "Even though the inner portion is brittle, the outer portion still is relatively tough because the radiation damage is attenuated through the wall," Cheverton said. "A crack might be driven through the inner part, but it tends to arrest at the outer part."

But that assessment could easily be wrong, says Pollard of the Union of Concerned Scientists. "There's no dispute that current emergency systems would not be able to cope with a fracture of the reactor vessel," he said. "For other problems, you can make a reasonable argument that you have some defense in depth. The defense-in-depth philosophy disappears when you talk about pressurized thermal shock."

The real problem, Pollard says, is that the nation's nuclear regulators and the manufacturers allowed a major construction program to roar ahead without considering the range of unknown dangers that lay before them.

"The Atomic Energy Commission went forward with all this undue optimism," complained Pollard, who resigned from his job as a regulator years ago in disgust. "Now we're in a position where nothing can be done to correct the mistakes without causing someone undue harm. I expected them to do the job back in the 1960s. Now everyone but the nuclear industry has to suffer."

"My perception is that the problem is well in hand," said Westinghouse's Sero. "We have significant research programs under way, we are putting significant money and engineering efforts into it, and we have a firm understanding that is going to improve, which will show that our predictions were very conservative." ■

REFERENCE 5

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

April 10, 1981

COPY FOR: ~~MR. MINOGUE~~

The Honorable Morris K. Udall
Chairman, Subcommittee on Energy
and the Environment
Committee on Interior and Insular Affairs
United States House Of Representatives
Washington, D. C. 20515

Dear Mr. Chairman:

On May 28, 1980 I wrote to you concerning the safety implications of control systems and dynamic characteristics of nuclear power plants. My comments then were intended to dispute the official NRC position that "safety systems will mitigate control system failures at any power".

One of the specific points I raised then, by way of an example of what Failure Mode and Effects Analyses (FMEA's) of control systems can and should uncover, was the likelihood of overcooling transients, generated by control system malfunctions in the secondary side of a Pressurized Water Reactor, as described in Reference 7 of that letter. Such transients can cause the reactor vessel to cool-down to about 150 °F in about 15 minutes, while the ECCS repressurizes it to about 2400 psi. This compound transient, known as pressurized thermal shock, is capable of catastrophically fracturing a reactor vessel that has been exposed to a neutron fluence corresponding to only a few Full Power Years Equivalent (FPYE) of operation, and has a high copper content of about 0.4% in its walls or welds.

A reactor vessel fracture is one of the most serious accidents a reactor may experience. Depending on its location and mode, it is almost certain that it will cause a core meltdown with all its public health and safety ramifications, on which, I am sure, I need not elaborate for you. Considering the high consequences of such an accident, then, one should ask what are the chances of it taking place. Unfortunately, such an accident is very likely and increasingly so. It is very likely because it may be caused by one or more failures in the non-safety control systems in the secondary side, and this is substantially supported by operational experience. It is increasingly so because as time goes on the neutron fluence to which the vessels of all reactors are exposed is increasing, and for several of them, I believe that a dangerous level has already been reached. I believe that this level is probably as low as 4 FPYE of operation for vessels with high copper alloy walls or welds. This is supported by analyses performed for the NRC, indicating that the overcooling transient that took place at Rancho Seco on March 20, 1978 would have caused such a vessel to rupture, had it been in operation for about 10 FPYE. However, that transient was not as severe as we can expect on a reasonable worst case basis. Furthermore, a recent discovery of a discrepancy existing

EXHIBIT
NO. 3

between the estimated vs. the measured values of neutron fluence for the Maine Yankee reactor vessel indicates a generic problem that makes things worse. The results of dosimetry measurements indicate the actual neutron fluence to be some 2.3 times higher than that estimated in the Maine Yankee Final Safety Analysis Report. Moreover, as you may recall, one of the measures ordered by NRC after the TMI-2 accident was to have all reactor operators not turn off the ECCS once it had been initiated. This might be desirable in some cases of accidents, but not necessarily in every case. For overcooling transients, without a large LOCA, the continued operation of ECCS compounds the accident by contributing to the cool-down process, and, most importantly, by repressurizing the primary system.

The pressurized thermal shock phenomena have not been the subject of experimental work by the NRC nor the industry. Nor have the control systems and their implications to safety been reviewed and analyzed. These crucial shortcomings pose some questions on the effectiveness of the regulatory process, which you may as easily as I ponder, but the immediate concern is to assure the safety of operating plants. Faced with the realities that we are faced today, and taking the approach that if we err, we should err in the direction of safety, it is apparent to me that those PWR's with high copper alloy vessels or welds, that have operated for 4 FPYE must be shutdown until this matter is resolved in the technical arena. It is conceivable that after additional and plant specific studies additional measures may be required.

Even though the Commission and the ACRS would probably respond to your letter of December 4, 1980 on the safety implications of control systems in a few months, I believe that this matter is serious and pressing enough, that requires a decision now. I believe that the Commission, with Congressional assistance and appreciation of the issues involved, will respond constructively.

If I can be of further assistance, please let me know.

Respectfully,

Demetrios L. Basdekas

Demetrios L. Basdekas
Reactor Safety Engineer

cc: Congressman Lujan
Congressman Markey
Chairman Hendrie

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

August 21, 1981

Docket No. 50-251

Dr. Robert E. Uhrig, Vice President
Advanced Systems and Technology
Florida Power and Light Company
P. O. Box 529100
Miami, Florida 33152

EXHIBIT
NO. 4

Dear Mr. Uhrig:

SUBJECT: PRESSURIZED THERMAL SHOCK TO REACTOR PRESSURE VESSELS

We have reviewed the PWR Owners' Groups responses of May 15, 1981 and the licensees' responses of May 22, 1981 to our letter dated April 20, 1981 concerning the subject issue. The EPRI work which bears on the issue was included in the licensees' responses. On the basis of our independent review, of the plants where neutron irradiation has significantly reduced the fracture toughness of the reactor pressure vessels (RPVs), all plants could survive a severe overcooling event for at least another year of full power operation. However, we believe that additional action should be taken now to resolve the long-term problems.

This belief is based upon our analyses which indicate that reductions in fracture toughness for some RPVs are approaching levels of concern. It is also based in part on the fact that any proposed corrective action must allow adequate lead time for planning, review, approval, procurement and installation. These conclusions were recently discussed with the PWR Owners Groups on July 28-30, 1981. At those meetings, the Owners Groups reviewed the programs underway at the three PWR vendors which are designed to scope the magnitude and applicability of the generic problem and to be completed by late 1981. The three programs appeared to contain the necessary elements for resolution of the problem on a generic basis and the NRC plans to make full use of the reports due by the end of the year. While the vendors and Owners Groups are to be commended and encouraged in addressing the generic issue, there is also a need for plant-specific information for your plant.

Based on current vessel reference temperature and/or system characteristics, we have identified Ft. Calhoun, Robinson 2, San Onofre 1, Maine Yankee, Oconee 1, Turkey Point 4, Calvert Cliffs 1 and Three Mile Island 1 as plants from which we require additional information at this time.

The staff has used the time-dependent pressure and temperature data from the March 20, 1978 Rancho Seco transient as a starting point for our evaluation of this issue because: (1) it is the most severe overcooling event experienced to date in an operating plant; (2) it is a real, as opposed to a postulated, event; and (3) it was severe enough that it could challenge the RPV when combined with physically reasonable values of irradiated fracture toughness and initial crack size. In future reviews the staff plans to use the steam line break accident or other appropriate transient/accident in order to estimate minimum operational times available before plant modifications are required.

Using calculated RPV steel mechanical properties, credible initial flaw sizes, reasonable thermal-hydraulic parameters, and a simplified pressure-temperature transient similar to that observed during the Rancho Seco event, the staff has concluded that all operating plants could safely survive such an event at the present time and for at least an additional year of full power operation. However, because of the required lead times for future actions, the margins in time for long term operation are not large, and there is considerable uncertainty in the probability that similar or more severe transients may occur. It is clear that positive action must be initiated soon for those plants with significantly high transition temperatures. As indicated above, several such plants have been selected by the staff, based on estimates of the current reference temperature for the nil ductility transition (RT_{NDT}) of the RPVs.

The need to initiate further action at this time is emphasized by the recognition that implementation of any proposed fixes or remedial actions must allow for adequate lead time. Because long-term solutions may require a year or more, you should explore short-term approaches as well. Although clear, concise instructions should be provided to operators to reduce the likelihood of repressurization during overcooling transients, the NRC staff believes that reliance on operator actions to prevent repressurization during an overcooling transient will be very difficult to justify as an acceptable long-term solution to the problem.

In accordance with 10 CFR 50.54(f) of the Commission's regulations, you are requested to submit written statements, signed under oath or affirmation, to enable the Commission to determine whether or not your license should be modified, suspended or revoked. Specifically, you are requested to submit the following information to the NRC within 60 days from the date of this letter:

(1) Provide the RT_{NDT} values of the critical welds and plates (or forgings) in your vessel for:
  (a) initial (as-built) conditions and location (e.g., 1/4 T) and
  (b) current conditions (include fluence level) at the RPV inside carbon steel surface.

(2) At what rate is RT_{NDT} increasing for these welds and plate material?

(3) What value of RT_{NDT} for the critical welds and plate material do you consider appropriate as a limit for continued operation?

(4) What is the basis for your proposed limit?

(5) Provide a listing of operator actions which are required for your plant to prevent pressurized thermal shock and to ensure vessel integrity. Include a description of the circumstances in which these operator actions are required to be taken. Included in this summary should be the specific pressure, temperature and level values for: a) high pressure injection (HPI) termination criteria presently used at your facility, b) HPI throttling criteria and instruction presently used at your facility and c) criteria for throttling feedwater presently used at your facility. For each required operator action, give the information available to the operator and the time available for his decision and the required action. State how each required operator action is incorporated in plant operating procedures and in training and requalification training programs.

You are also requested to submit a plan for Turkey Point, Unit No. 4 to the NRC within 150 days of the date of this letter that will define actions and schedules for resolution of this issue and analyses supporting continued operation. We request that you include consideration and evaluation of the following possible actions:

(1) reduction of further neutron radiation damage at the beltline by replacement of outer fuel assemblies with dummy assemblies or other fuel management changes;

(2) reduction of the thermal shock severity by increasing the ECC water temperature;

(3) recovery of RPV toughness by in-place annealing (include the basis for demonstrating that your plant meets the requirements in 10 CFR 50 Appendix G IV C);

(4) design of a control system to mitigate the initial thermal shock and control repressurization.

For these, as well as for any other alternative approaches, provide implementation schedules that would assure continuance of adequate safety margins.

In the interest of efficient evaluation of your submittal, we request that you include with the above plan, a response to the enclosed request for additional information.

Due to the nature of this review, and the past review effort that has been expended, we consider the above schedules to be reasonable; however, inform us within 30 days if you anticipate conflicts with previous commitments with either submittal and a basis for any delay. We also expect participation by the appropriate PWR Owners Group and NSSS vendors in developing solutions to the problem.

Sincerely,

for Darrell G. Eisenhut, Director
Division of Licensing
Office of Nuclear Reactor Regulation

Enclosure:
Request for Additional Information

cc w/enclosure:
See next page

P.O. BOX 529100 MIAMI, FL 33152

FPL

FLORIDA POWER & LIGHT COMPANY

October 23, 1981
L-81-465

Office of Nuclear Reactor Regulation
Attention: Mr. Darrell G. Eisenhut, Director
Division of Licensing
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

RECEIVED
OCT 28 1981
U.S. NUCLEAR REGULATORY COMMISSION

Dear Mr. Eisenhut:

Re: FPL Letter L-81-462
Dated October 21, 1981
ATTACHMENT CORRECTION

By our letter referenced above, we responded to questions of your letter dated August 21, 1981 regarding Turkey Point Units 3 & 4 relating to pressurized thermal shock to reactor pressure vessels.

The attached corrected page replaces Attachment Page 1 of our letter referenced above.

Very truly yours,

Jack E. Vessely for

Robert E. Uhrig
Vice President
Advanced Systems & Technology

REU:DAC:cf

Attachment

cc: Mr. James P. O'Reilly, Region II
Mr. Harold F. Reis, Esquire

EXHIBIT
NO. 275

A049
S/1

285
P

Pressurized Thermal Shock to Reactor Pressure Vessels

Question (1):

Provide the RT_{NDT} Values of the critical welds and plates (or forgings) in your vessel for:

(a) Initial (as built) conditions and location (e.g. 1/4T) and
(b) current conditions (include fluence level) at the RPV inside carbon steel surface.

Response (1):

		Material	Initial RT_{NDT}	RT_{NDT}^{+}	Current RT_{NDT}
(a)	Intermediate Inner Forging*	123P481VA1	+50 F	+ 35 F	+ 85F
	Circumferential (Girth) Weld**	SA 1101	+ 3 F	+190 F	+193F
	Lower Forging*	122S180VA1	+40 F	+ 35 F	+ 75F

* 1/4 T

** Inner wall. The current RT_{NDT} (1/4 T) = +168 F. Value is based on Unit 3 data which has been shown to be more representative of Unit 4 than surveillance capsule removed from Unit 4 (L-77-113, dated April 11, 1977 and L-77-326, dated October 21, 1977).

\+ Based on the slope of prediction curves presented in proposed ASTM Standards "Predicting Neutron Radiation Damage To Reactor Vessel Material."

(b) There have been 5.61 Effective Full Power Years (EFPY) of operation as of September 30, 1981 at Turkey Point Unit 4.

The total fluence on the inner wall is 1.1×10^{19} n/cm^2 and 6.6×10^{18} n/cm^2 at 1/4 T.

Question (2):

At what rate is RT_{NDT} increasing for these welds and plate material?

Response (2):

RT_{NDT} is increasing at the rate of 7°F/EFPY for the next 10 years; for the remainder of life, 5°F/EFPY. The rate of change for the forgings is 3°F for the remaining design life of the vessel. These are based on the slope of prediction curves presented in proposed ASTM Standards "Predicting Neutron Radiation Damage To Reactor Vessel Material."

Question (3):

Provide a listing of operator actions which are required for your plant to prevent pressurized thermal shock and to ensure vessel integrity. (Include

WCAP 8631

ANALYSIS OF CAPSULE T FROM THE FLORIDA POWER AND LIGHT COMPANY TURKEY POINT UNIT NO. 3 REACTOR VESSEL RADIATION SURVEILLANCE PROGRAM

S. E. Yanichko
J. H. Phillips
S. L. Anderson

December 1975

Work performed under Shop Order No. MIP-23572

APPROVED: J. N. Chirigos, Manager
Structural Materials Engineering

WESTINGHOUSE ELECTRIC CORPORATION
Nuclear Energy Systems
P. O. Box 355
Pittsburgh, Pennsylvania 15230

SECTION 2

INTRODUCTION

This report presents the results of the examination of Capsule T, the first capsule of the continuing surveillance program which monitors the effects of neutron irradiation on the Florida Power & Light Company, Turkey Point Unit No. 3, reactor pressure vessel materials under actual operating conditions.

The surveillance program for the Turkey Point Unit No. 3 reactor pressure vessel materials was designed and recommended by the Westinghouse Electric Corporation. A description of the surveillance program and the preirradiation mechanical properties of the reactor vessel materials are presented in WCAP-7656.[1] The surveillance program was planned to cover the 40-year life of the reactor pressure vessel and was based on ASTM E-185-66, "Recommended Practice for Surveillance Tests on Structural Materials in Nuclear Reactors."[2] Westinghouse Nuclear Energy Systems personnel were contracted for the preparation of procedures for removing the first capsule from the reactor and its shipment to the Westinghouse Research and Development Laboratory, where the post irradiation mechanical testing of the Charpy V-notch impact, tension, and Wedge Opening Loading (WOL) surveillance specimens was performed.

This report summarizes testing and the post irradiation data obtained from the first material surveillance capsule (Capsule T) removed from the Turkey Point Unit No. 3 reactor vessel and discusses the analysis of these data.

SECTION 1
SUMMARY OF RESULTS

The analysis of the reactor vessel material contained in the first surveillance capsule from the Florida Power and Light Company, Turkey Point Unit No. 3, reactor pressure vessel led to the following conclusions:

- The capsule received an average fast fluence of 5.58×10^{18} neutrons/cm^2 (E > 1 Mev). The predicted fast fluence for the capsule at the end of first core cycle was 5.27×10^{18} neutrons/cm^2 (E > 1 Mev).

- The fast fluence of 5.58×10^{18} n/cm^2 (E > 1 Mev) resulted in a 190°F increase in the 50 ft lb reference nil-ductility transition temperature (RT_{NDT}) of the weld metal, which is the most limiting material in the core region of the reactor vessel. The intermediate pressure vessel shell forging (123P461VA-1) exhibited essentially a 0°F shift in the 50 ft lb nil-ductility transition temperature (specimens oriented in the major working direction of the forgings). The weld heat-affected zone material also exhibited a 0°F shift in transition temperature.

- Based on a ratio of 2.48 between the fast flux at the surveillance capsule location to that at the vessel wall and an 80 percent load factor, the projected fast fluence which the Turkey Point Unit No. 3 reactor pressure vessel will receive after 40 calendar year operation is 6.65×10^{19} n/cm^2 (E > 1 Mev). This fluence is approximately the same as the 6.30×10^{19} n/cm fluence calculated for 40 year operation.

- The projected shift in RT_{NDT} of the weld metal after 40-calendar-year operation is 382° and 330°F at the vessel inside surface and the 1/4 thickness location respectively.

- The average upper shelf impact energy of the weld metal decreased from 64.5 to 58.5 ft lb during the first core cycle.

- The irradiated properties of forging 123P461VA-1 and the weld metal are adequate to provide for continued safe operation of the Turkey Point Unit No. 3 power plant.

- The fracture toughness as measured dynamically on WOL specimens gave results well above the limits of the K_{IR} curve in Appendix G of ASME Code Section III.

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San Antonio, Texas 78284

ACTOR VESSEL MATERIAL SURVEILLANCE PROGRAM
FOR
TURKEY POINT UNIT NO.4
ANALYSIS OF CAPSULE T

by

E. B. Norris

FINAL REPORT
SwRI Project No. 02-4221

to

Florida Power & Light Company
P. O. Box 3100
Miami, Florida 33101

June 14, 1976

Approved:

U. S. Lindholm, Director
Department of Materials Sciences

Location in Wall	Lead Factor	Shift in RT_{NDT} (deg F)			
		3 EFPY	5 EFPY	10 EFPY	32 EFPY
1/4T	4.17	242	281	342	467
3/4T	17.4	162	188	230	312

These values were used as the bases for computing heatup and cooldown limit curves for Turkey Point Unit No. 4. (Three EFPY will not be reached until near the end of Core Cycle IV as estimated from both computer predictions and past operating experience.)

(9) Assuming that the percent change in Charpy V-notch upper shelf energy is proportional to the square root of the neutron fluence, the weld metal upper shelf energy at the 1/4T position is predicted to reach the 50 ft-lb level at approximately 2.7 EFPY of operation.

(10) Although the surveillance program is in compliance with Appendix H of 10CFR50, it is recommended that a replacement capsule with additional weld metal specimens be placed in the Capsule T slot if archival material is available. An alternative is to move Capsule V into the Capsule T slot at the end of Core Cycle III (April 1977) and remove it for testing at the end of Core Cycle IV (April 1978), at which time the estimated fluence on Capsule V would be 8.25×10^{18} neutrons per cm^2 ($E > 1$ MeV).

(11) On the basis of NRC recommendations, the WOL fracture mechanics specimens have been stored untested pending development of recommendations concerning test procedures.

of nearly 1×10^{19}. Therefore, the projections of shift in RT_{NDT} have been based on the Westinghouse curves. The result obtained from Capsule T has been added to Figure 9, and a normalized response curve has been drawn through the data point parallel to the Westinghouse curves. The predicted shifts in RT_{NDT} for the Turkey Point Unit No. 4 reactor pressure vessel obtained from Figure 9 are summarized in Table XI. The values predicted at the 1/4 T and 3/4 T are used to develop heatup and cooldown limit curves to meet the requirements of Appendix G to Section III of the ASME Code.

These projections for C_v shelf energy reductions and RT_{NDT} shifts are based on one data point, Capsule T, and trend curves for like materials. It is anticipated that the reliability of the trend curves will be improved as more surveillance data becomes available and a better understanding of the factors affecting radiation embrittlement has been achieved. As an example of the latter, Mr. E. C. Biemiller of Combustion Engineering, in a paper given at the ASTM E10 Symposium on Effects of Radiation on Structural Materials in St. Louis, May 4-6, indicated that a parameter of (% Ni + % Si) + (% Mo + % Cr + % Mn) may explain the variation in radiation embrittlement observed in ferritic materials of nominally the same copper content. Also, the Metal Properties Council is developing new radiation damage curves that will be based on more data than those currently in use.

Because of the potential of reaching a low C_v shelf energy condition in the Turkey Point Unit No. 4 weld metal in the next few years, it is advisable to obtain another data point in the not too distant future.

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PRESSURE-TEMPERATURE LIMITATIONS FOR THE TURKEY POINT UNIT NOS. 3 AND 4 NUCLEAR POWER PLANTS

by
E. B. Norris
J. F. Unruh

SwRI Project No. 02-4383-039

to

Florida Power and Light Company
Miami, Florida

June 30 1976

EXHIBIT
NO. 8

Approved:

U. S. Lindholm, Director
Department of Materials Sciences

The first surveillance capsule was removed from Unit No. 4 during the 1975 refuelling outage. This capsule (also identified as "T") was evaluated by SwRI, and the results have also been reported.* The Unit No. 4 weld metal was also found to be the limiting material for controlling the vessel RT_{NDT}, and it exhibited an even greater sensitivity to neutron radiation embrittlement.

As a part of their analysis of Capsule T, Westinghouse computed heatup and cooldown limit curves for Unit No. 3 for 3 and 10 effective full power years. In their analysis, they employed additional conservatism above ASME Section III Code requirements by applying a 1.25 safety factor to the stress intensity factor caused by thermal gradients. Florida Power & Light Company asked SwRI to recompute the Unit No. 3 heatup and cooldown limit curves and compute heatup and cooldown limit curves for Unit No. 4 using the safety factors called out in Appendix G of Section III of the ASME Code.

B. Input Information

1. Fracture Toughness Properties

The values of RT_{NDT} for the beltline regions of Turkey Point Unit Nos. 3 and 4 were derived from (1) the surveillance program test results, (2) computed ratios of fast flux at the capsule location to the maximum fast flux at the 1/4T and 3/4T locations in the vessel wall, and

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* Norris, E. B., "Reactor Vessel Material Surveillance Program for Turkey Point Unit No. 4 - Analysis of Capsule T," Final Report, SwRI Project 02-4221, June 14, 1976.

(3) trend curves of increase in RT_{NDT} as a function of neutron fluence ($E > 1$ MeV). A summary of these values is as follows:

Unit No.	Operating Period*	RT_{NDT} at 1/4T	RT_{NDT} at 3/4T
3	5 EFPY	194°F	131°F
3	10 EFPY	236°F	159°F
4	5 EFPY	281°F	188°F
4	10 EFPY	342°F	230°F

* EFPY = Effective Full Power Year

2. **Vessel Constants**

The following input data were employed in this analysis:

Inner Radius, r_i	=	77.75 in.
Outer Radius, r_o	=	85.78 in.
Operating Pressure, P_o	=	2235 psig
Initial Temperature, T_o	=	70°F
Final Temperature, T_f	=	550°F
Effective Coolant Flow Rate, Q	=	97×10^6 lb_m/hr
Effective Flow Area, A	=	19.15 ft^2
Effective Hydraulic Diameter, D	=	11.9 in.

C. **Heatup and Cooldown Limit Curves**

Heatup curves were computed for a heatup rate of 100°F/hr. Since lower rates tend to raise the curve in the central region (see Figure 8), these curves apply to all heating rates up to 100°F/hr. Cooldown curves were computed for cooldown rates of 0°F/hr (steady state), 20°F/hr, 60°F/hr and 100°F/hr. The 20°F/hr curve would apply to cooldown rates up to 20°F/hr; the 60°F/hr curve would apply to rates from 20°F to 60°F/hr; the 100°F/hr curve would apply to rates from 60°F/hr to 100°F/hr.

The Unit No. 3 heatup and cooldown curves for up to 5 EFPY are given in Figures 10 and 11. Unit No. 3 curves covering 5 to 10 EFPY are given in Figures 12 and 13. Corresponding curves for Unit No. 4 are given in Figures 14 through 17.

NUREG/CR-2837

PNL Technical Review of Pressurized Thermal Shock Issues

Manuscript Completed: June 1982
Date Published: July 1982

Prepared by
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Pacific Northwest Laboratory
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Prepared for
Division of Safety Technology
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555
NRC FIN B2510

ABSTRACT

Pacific Northwest Laboratory (PNL) was asked to develop and recommend a regulatory position that the Nuclear Regulatory Commission (NRC) should adopt regarding the ability of reactor pressure vessels to withstand the effects of pressurized thermal shock (PTS). Licensees of eight pressurized water reactors provided NRC with estimates of remaining effective full power years before corrective actions would be required to prevent an unsafe operating condition. PNL reviewed these responses and the results of supporting research and concluded that none of the eight reactors would undergo vessel failure from a PTS event before several more years of operation. Operator actions, however, were often required to terminate a PTS event before it deteriorated to the point where failure could occur. Therefore, the near-term (less than one year) recommendation is to upgrade, on a site-specific basis, operational procedures, training, and control room instrumentation. Also, uniform criteria should be developed by NRC for use during future licensee analyses. Finally, it was recommended that NRC upgrade nondestructive inspection techniques used during vessel examinations and become more involved in the evaluation of annealing requirements.

1.0 INTRODUCTION

1.1 BACKGROUND

The pressure vessel of a nuclear plant is subjected to a pressurized thermal shock (PTS) when an extended cooling transient to the vessel wall is accompanied by system pressurization. Under these conditions, thermal and pressurization stresses on the internal surfaces of the vessel are additive. Moreover, these stresses are in tension and tend to open cracks located at or near the internal surfaces.

Nuclear plant pressure vessels are fabricated from ferritic steels. The internal surfaces of the vessels are clad with stainless steel weld to prevent metal corrosion processes. The vessels are designed to withstand normal heating and cooling transients for the life of the plant, which is usually 40 years at 80% operating efficiency or 32 effective full-power years (EFPY). A pressure vessel intended for 32 EFPY must be designed to maintain fracture toughness of the vessel material. An adequate level of fracture toughness provides assurance that small cracks will not propagate in a "brittle" manner as a result of stresses during an abnormal transient such as a PTS event. Failure in a brittle manner could fracture the vessel wall and lead to severe failure of the pressure boundary in the core area. In contrast, a ductile type of failure would be expected to result, at worst, in a through-vessel crack, which would leak but not result in a total loss of the pressure boundary.

In older nuclear plants, the pressure vessels were often fabricated with weld materials containing relatively high levels of copper, phosphorus, and nickel. These elements were later shown to result in greater irradiation damage to the vessel material than had been initially expected. Irradiation damage caused a shift in the fracture toughness curve to higher temperatures and, therefore, increased the remote possibility of a nonductile vessel failure.

Evaluating the failure probability of any nuclear pressure vessel is very complex. The evaluation must be plant-specific to allow for differences in material properties of the plant components, systems configuration, operating procedures, and dosimetry history. The plant control systems, component redundancy, operating history, and operator training and proficiency are important in determining the initiation, sequence, and timing of accident-type events and in evaluating the probability of mitigating operator actions. Finally, the thermal-hydraulic, material properties, and fracture mechanics analyses, using currently available codes, are used to determine the consequences of the events being analyzed.

The following conditions must be present during a PTS event before a significant nonductile failure probability would be expected:

- The nuclear plant pressure vessel must exhibit significant loss of fracture toughness through neutron irradiation.

- An overcooling transient must occur that would be of sufficient duration to cause a steep thermal gradient across the vessel wall and cooling to the low-toughness temperature range.

- A flaw must be present of sufficient size and be located at a critical beltline location where reduced fracture toughness and high thermal stress exist.

- A simultaneous high reactor coolant system pressure must be present.

In recent years a number of incidents have occurred that involved several, but not all, of the above conditions. The PTS issue is, therefore, being investigated in much greater detail by the NRC, the utility industry, and Nuclear Steam Supply System (NSSS) contractors.

1.2 OBJECTIVE OF STUDY

Pacific Northwest Laboratory is providing technical assistance to NRC to develop and recommend a regulatory position that NRC should adopt before the longer-term PTS program provides generic resolution and acceptance criteria. The near-term recommendations include any corrective actions required at the eight plants identified in the August 21, 1981 NRC letter.(1) The recommendations of this report are based on the review of information described in Section 1.3.

1.3 APPROACH

Eight pressurized water nuclear power plants (Ft. Calhoun, H. B. Robinson 2, San Onofre 1, Maine Yankee, Oconee 1, Turkey Point 4, Calvert Cliffs 1, and Three-Mile Island 1) have been identified for specific review of PTS event scenarios. These plants and the NSSS owners groups have supplied information in response to NRC requests.(2,3,4) The following sources of information were used by PNL to recommend NRC's near-term regulatory position.

1. Documentation by the licensees and owner groups to the NRC requests for information concerning the PTS issue.

2. Participation in reviewing current procedures, training, and operator responses to PTS events at selected plants as established by the NRC's PTS task force on procedure review.

3. Reviews of research work being performed in support of the PTS issue at NRC, national laboratories, industry, and other research institutes.

5.0 MATERIALS PROPERTIES OF IRRADIATED VESSELS

Pressure vessel steels exposed to neutron irradiation experience a degradation in fracture resistance. Ferritic steels have an intrinsically poor fracture resistance at low temperatures. The loss of ductility with decreasing temperature occurs as the nil-ductility transition temperature is approached. Below the transition temperature materials fail by unstable, brittle fracture, whereas above that temperature materials fail by stable, ductile fracture. Neutron irradiation causes the nil-ductility transition reference temperature (RT_{NDT}) to shift to higher temperatures. The shift can be large enough to endanger the integrity of the pressure vessel if the irradiation-shifted nil-ductility temperature is elevated above the service temperature of the vessel wall. Of particular concern is the fracture resistance of irradiation-sensitive welds.

Two factors aggravate the fracture resistance of irradiated vessel welds subjected to a PTS event. In some cases, aggravation occurs when the irradiation history of the reactor has resulted in significant elevation of the nil-ductility temperature. In other cases, aggravation occurs when PTS lowers the wall temperature, which thus lowers the fracture resistance of the vessel welds. Accurately predicting the fracture of a vessel weld requires estimating the vessel neutron exposure histories, welding procedures, and the irradiation sensitivities of welds as a function of chemistry. Furthermore, the radial dependence of neutron spectrum and flux in the wall must be evaluated to quantitatively determine the increasing fracture toughness through the wall.

This chapter describes the effects that irradiation and material characteristics have on the degraded fracture resistance of pressure vessel steels. Methods used by licensees and owners groups to predict fracture resistance, and the uncertainties inherent in these methods, are evaluated. Lastly, the state of knowledge is evaluated to indicate what information may become available in the future which would aid in evaluating the integrity of irradiated pressure vessels during a PTS event.

5.1 NEUTRON DOSIMETRY

Atomic displacements caused by neutron irradiation are the principal cause of degraded fracture toughness of nuclear pressure vessel steels. The degradation is directly related to the number of high-energy neutrons that penetrate the steel. Traditionally, the number of neutrons having an energy greater than 1 MeV has been used to characterize the irradiation exposure. Predicting the material properties of plant-specific reactor vessels requires an accurate knowledge of neutron exposures of metallurgical test specimens and an accurate knowledge of the neutron exposure of plant-specific pressure vessel components.

Methods used to irradiate and test metallurgical specimens and to estimate neutron exposure of vessel components result in uncertainties that affect the predicted reliability of vessels during a PTS event. Accurately defining the neutron irradiation environment requires knowledge of the neutron spectra, flux, and fluence, as well as the irradiation temperature. Irradiation of surveillance specimens provides the most reliable data base for predicting the irradiation properties of vessel components. Such data have the most credibility, because they most accurately represent the neutron environment inside a vessel wall. The plant-specific neutron spectra and fluxes are similar for surveillance irradiations and inner-wall vessel irradiations.

Methods used for vessel dosimetry are dependent on dosimetry analyses of surveillance capsules and on calculated neutron fluxes. Discrete Ordinate Transport (DOT) codes are used by the licensees and owners groups to map out the spatial dependence of neutron flux. The calculated fluxes are then compared with measured fluxes using flux monitors inserted in surveillance capsules. The DOT codes are considered to be accurate, but if wrong input values are assumed, the predicted fluxes can be inaccurate. When predicted fluxes are compared with measured fluxes, the values can agree to within 10% to 15%.[14] The uncertainty in peak fluence values provided by the licensees and owners groups is reasonable; the values for Combustion Engineering were within 30%, the values for Westinghouse were within 20%, and the values for Babcock & Wilcox were approximately 15%. The discrepancies in peak fluence values represent uncertainty in the predicted peak fluence ($E > 1$ MeV) at the inner surface of the steel vessel.

Additional uncertainty can exist in the predicted vessel properties because irradiation tests and vessel walls have different neutron spectra and fluxes. These differences are minimized when the properties of surveillance specimen are correlated to vessel properties. The correlation is possible because the neutron spectrum and flux of the surveillance location are similar to those found inside the vessel wall. When projecting properties through the thickness of the vessel wall, the spectrum and flux are degraded. The spectrum is shifted toward a lower average energy with many neutrons below 1 MeV contributing to irradiation damage.

To account for these lower energy neutrons, it has been recommended that displacements per atom (DPA) be used as a measure of irradiation exposure. The damage based on DPA is greater through the wall than would be predicted based on the $E > 1$ MeV assumption. Differences between the two exposure criteria as a function of distance through a vessel wall are given in Table 5.1.[15]

As radial distance increases, damage rates decrease. The lower damage rates may provide a greater opportunity for self annealing during irradiation. Hence, damage accumulates more slowly per DPA for positions deep in a vessel wall. This suggests a lesser damage in deep regions than would be expected if rate effects on damage efficiency were neglected when predicting radiation damage through a vessel wall. The effect of the damage rate efficiency can be estimated by comparing damage rates with thermal annealing rates. Combinations

NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

December 3, 1982

MEMORANDUM FOR: Commissioner Gilinsky
Commissioner Ahearne

FROM: Demetrios L. Basdekas
Instrumentation & Control Branch
Division of Facility Operations
Office of Nuclear Regulatory Research

SUBJECT: STAFF REPORT ON PRESSURIZED THERMAL SHOCK, SECY-82-465, NOVEMBER 23, 1982.

Earlier this week I discussed with you a number of, what I considered to be, significant points on the subject staff report. Unfortunately, access to the report was denied to me until the afternoon of November 29, 1982. Based on my limited review of this final version, I have prepared the following summary of the points I discussed with you including a few additional ones.

1. The probability of a PTS caused vessel rupture and core-melt is not quantifiable with a certainty which can form the primary basis for decision-making on this matter. I believe the heavy reliance on the quantitative risk estimates by the staff is unwarranted. I also believe that the staff should provide a complete and thorough response to the two basic questions:

   (a) What is the uncertainty in the estimate of the probability of PTS-caused catastrophic vessel rupture and core melt?

   (b) What is the confidence level in that uncertainty? How was it derived? (See pp. 8-9, H-26, Sec. H.4.1 of Staff report)

2. The Rancho Seco event was considered by the Staff to be the most severe, and so stated on p. H-26, first full paragraph of the main report. The Crystal River event as described in Section 2.2.6 and as shown on Figure 2-13 of the main report appears to be more severe than the Rancho Seco event. I believe that the Staff should provide a definition of severity and answers to the following questions:

   (a) Is the Rancho Seco or the Crystal River event the most severe?

   (b) How confident are we on the time history data of temperature and pressure that have been provided by the licensees?

   (c) If Rancho Seco is not the most severe event, how does this affect the analyses performed on the assumption that it was?

EXHIBIT
NO. 16

3. In-situ annealing capability of PWR vessels has not been demonstrated and there is considerable doubt that it will be available for a long time, if ever, during the lifetime of most PWRs which have exceeded or are expected to exceed 200°F of their vessel RT_{NDT}. There are also plants that are facing similar difficulties with regard to an acceptable limit of the upper shelf toughness. I believe that the following question should be answered in some detail:

   (a) Which plants do not meet existing regulations? (i.e., Appendix G, IV, A2a, B, and C relating to upper shelf toughness and in-situ annealing capability).

4. The staff acknowledges the importance of Instrumentation and Control Systems malfunctions in PTS (See second full paragraph beginning on p. 1 of SECY-82-465), but it has not asked the utilities to supply design information on these systems and their electrical power supplies in its letters of August 1981 and since then (second full paragraph on page 3 of SECY-82-465). The following questions have been asked by the Commissioners before, in one form or another, but no definitive answer has been provided, to my knowledge.

   (a) What is the reason for this inconsistency between the stated importance of instrumentation and control systems (p. 1) and stated actions (p. 3)?

   (b) If we do not have a timely and technically sound resolution of USI A-47, Safety Implications of Control Systems, how can you expect to resolve A-49, PTS?

   Furthermore,

   (c) Without design information on the plants we have chosen to review under both USIs A-47 and A-49 (Oconee-1, Calvert Cliffs-1, H. P. Robinson-2) how can we justify the large expenditures of our RES and NRR programs which deal with I&C systems initiated transients of importance to PTS or any other safety issue?

5. The proposed screening criteria are 270°F for longitudinal and 300°F for circumferential welds in the RPVs. The selection of the screening criteria method is based on eight events taken on a cumulative manner of <u>all</u> PWR experience. This leaves out probability components associated with (a) substantial operational experience involving event sequences which terminated early enough or in some other benign way, which might, with some probability, have continued on to produce a more severe challenge to the RPV and (b) essentially those potential events and their associated sequences, which have not occurred yet, but which may, with some probability, occur in the future, causing a severe challenge to the RPV. These are important considerations in estimating probabilities of event sequences that

TABLE 5.2. Summary of Weld Properties and RT_{NDT} Predictions

Plant	Weld Location	RT^0_{NDT}, °F	Fluence, n/cm^2	Date	Cu, %	Ni, %	$RT^{(a)}_{NDT}$ Mean + 2σ
Turkey Pt. 4	Circum.	+20	1.10×10^{19}	9/30/81	0.30	0.57	265
Fort Calhoun	Long 2-410	-20	6.48×10^{18}	12/31/81	0.35	0.99	268
San Onofre 1	Long 7-860A	-20	2.75×10^{19}	10/31/81	0.35	0.20	278
Calvert Cliffs 1	Long 2-203	-20	7.05×10^{18}	12/31/81	0.30	0.99	267
Maine Yankee	Long 2-203	-20	4.73×10^{18}	12/31/81	0.36	0.99	251
Robinson 2	Long 2-273	-20	1.30×10^{19}	9/30/81?	0.34	0.20	218
	Circ 11-273	-20	1.24×10^{19}	(assumed)	0.34	0.50	253
Oconee 1	Long 3A-1430	+20	2.27×10^{18}	10/01/81	0.31	0.55	183

(a) $RT_{NDT}\,(HEDL) = RT^0_{NDT} + ([illegible]8 + 470*Cu + 350*Cu*Ni)*\left(\frac{F}{1 \times 10^{19}}\right)^{0.27}$.

5.3 IRRADIATION PROPERTIES

The shift in the nil-ductility temperature due to neutron irradiation of pressure vessel steels is well known. The issue for the PTS evaluation is to quantify the irradiation shift as accurately as possible for specific vessel welds. Because specimens cannot be extracted from the irradiated vessels, it is necessary to project irradiation properties from irradiations of metallurgical test specimens. The irradiation environment and materials used for these metallurgical specimen irradiations must approximate, as much as possible, the materials and environment of the pressure vessel. Furthermore, irradiation tests must project the properties at some future date--in particular, to end of life or 32 EFPY.

The irradiation tests that were used to establish Regulatory Guide 1.99, Rev. 1 were performed primarily in test reactors at enhanced fluxes and in neutron spectra having average energies larger than those typical for pressure vessels. The rapid fluxes meant that fluences in end-of-life reactor vessels may cause a PTS of certain severity in the future. Hence, plant-specific analyses are needed to estimate a meaningful number of probability and RT_{NDT} screening value. I believe we should consider a criterion for each vendor design and ultimately a limit for each plant. (See Comment No. 6, below). These limitations are acknowledged by the staff but their significance apparently is somewhat elusive when it comes to formulating the conclusions and recommendations for the screening criteria.

An important question that should not escape serious consideration is:

(a) How do we reconcile this selection of screening criteria with the fact that a Small Break LOCA is capable of cooling down the vessel to about 125°F within about 30 minutes with a subsequent isolation and repressurization to full design pressure?

6. The summary of operational experience given in Section 2.3, Figure 2-14, Figure 4-1, and elsewhere in the Staff report, provides a lumping of the operational experience for reactors designed by all three vendors. This results in a "smearing" or "averaging out" of the operational data associated with reactors of individual vendors. I believe that a meaningful PTS assessment may be performed on a plant-specific basis only, and with substantial limitations on a vendor-generic basis, but, I believe, with almost nil utility for an all-vendor-generic basis. Hence, the selection of the screening criteria discussed in Chapter 4.0 is based on very weak grounds. (See Comment No. 5, above)

I believe that the following question should be answered by the Staff:

(a) Would you explain how, in your judgment, the lumping together of operational experience from plants supplied by all three NSSS vendors (B&W, W, and CE) gives you a realistic and applicable data base for all of them, when you consider the fact that the dominant contribution comes from B&W plants PTS precursor events?

(b) What is the combined effect on your selection of screening criteria when you take into account the consideration of the points discussed in Comments 5 and 6?

7. A flux reduction by a factor of 2-4.5 gives a ΔRT_{NDT} of [illegible]. This is well within the uncertainty band of the estimated RT_{NDT}.

   (a) Does the Staff think that a more meaningful and prudent flux reduction would be one by a factor of 10-30 for plants that flux reduction is needed?

8. In-service inspection techniques and frequency requirements are not very effective in producing useful and timely data for use in PTS related analyses.

   I believe that the Commission may find it appropriate to request the Staff to address this issue in some detail (including Staff members' opinions, which may be in variance with the position stated by the Staff during the December 1, 1982 briefing).

The enclosed memorandum (1) contains my comments on the September 13, 1982 draft of the subject Staff report. A number of them have been taken into propper consideration in the final report. However, the most important ones have remained unresolved.

My only recommendation at this time is that the Commission consider the following interim screening criteria for both longitudinal and circumferential welds or bulk plate materials wherever they may be governing:

For Babcock and Wilcox Plants: RT_{NDT} of 150°F

For Westinghouse and Combustion Engineering Plants: RT_{NDT} of 200°F

These screening criteria would provide for more realistic, timely, and prudent resolution of this issue.

Dr. Okrent's recommendation (2) for the Commission's active participation in "establishing the criteria to be used on this issue for decision making under uncertainty" (emphasis added) is very appropriate.

I appreciate the opportunity to have discussed with you most of my comments made above, and I will be pleased to answer any questions you may have on them as well as my recommendations for interim screening criteria.

By copy of this memorandum I am confirming my pending requests to your colleagues on the Commission for the opportunity to meet and discuss with them individually my views on this important issue.

Demetrios L. Basdekas

Demetrios L. Basdekas
Instrumentation and Control Branch
Division of Facility Operations
Office of Nuclear Regulatory Research

Enclosures/References:

1. Memorandum from D. L. Basdekas to P. S. Shewmon, ACRS, October 6, 1982. (Enclosure)

2. Letter from P. S. Shewmon, ACRS to Chairman Palladino, October 14, 1982 - Additional Comments by ACRS Member David Okrent (Reference)

cc: Chairman Palladino
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Commissioner Asselstine
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G. C. Sih
Director

October 10, 1985

Attorney Martin H. Hodder
1131 N.E. 86th Street
Miami, Florida 33138

RE: Turkey Point Nuclear Power Plant Unit No. 4: Reactor Vessel Embrittlement and Surveillance Program

Dear Attorney Hodder:

In response to your letter dated August 29, 1985 and the above referenced subject matter, I have read the package of documents on the RPV embrittlement program at Turkey Point Unit No. 4. A number of supporting arguments with reference to the calculation of ΔRT_{NDT} are questionable, if not invalid from the scientific view point. In what follows, the SWRI report and the FPL letter shall be referred to as [1]* and [2]**, respectively.

(1) SWRI Prediction [1]

Based on the RPV material surveillance methodology, SWRI [1] estimated the shift in RT_{NDT} for Turkey Point Unit No. 4. The results pertaining to wall location 1/4T based on the data of Capsule T in terms of EFPY are summarized graphically on the sheet attached to this letter. The shift in RT_{NDT} is found to be approximately 324°F at 8 EFPY. This is beyond the NRC screening value of 300°F.

* E. B. Norris, "Reactor Vessel Material Surveillance Program for Turkey Point Unit No. 4: Analysis of Capsule T", Southwest Research Institute Technical Report No. 02-4221, June 1976.

** Letter, Uhrig, FPL, to Eienhut, "Re: Turkey Point Unit 4, Docket Nos. 50-251, PTS to Reactor Pressure Vessels", January 21, 1982.

EXHIBIT
NO. 11

(2) FPL Response [2]

With reference to the material in Docket No. 50-251 on PTS of RPV as stated in [2], a lower ΔRT_{NDT} value of 211°F was obtained for Unit No. 4. This result, however, was obtained by application of the surveillance data taken from Turkey Point Unit No. 3. The justification was that the metallurgical properties of the beltline welds of the Turkey Points Units No. 3 and No. 4 are the same and that data on Unit No. 4 are not sufficient.

(3) Comments

The rate at which the beltline weld material deteriorates and/or embrittles depends on the *combined* effects of irradiation and pressurized thermal shock. It is plant-specific in the sense that the influence differs inherently from one unit to another. In other words, the metallurgical properties alone cannot determine the damage behavior of the welds. The *loading history* plays a major role. Unless the rates of irradiation, fluctuations in thermal gradients and time variation in pressure are exactly the same for both Units No. 3 and No. 4, one is not justified to assume that data collected in Unit No. 3 could be applied to predict the behavior of Unit No. 4. Hence, conclusions drawn on ΔRT_{NDT} for Unit No. 4 based on the data of Unit No. 3 cannot be considered valid.

I will not delve into the other details concerning the actual calculation of ΔRT_{NDT} as they are beyond the scope of our immediate concern.

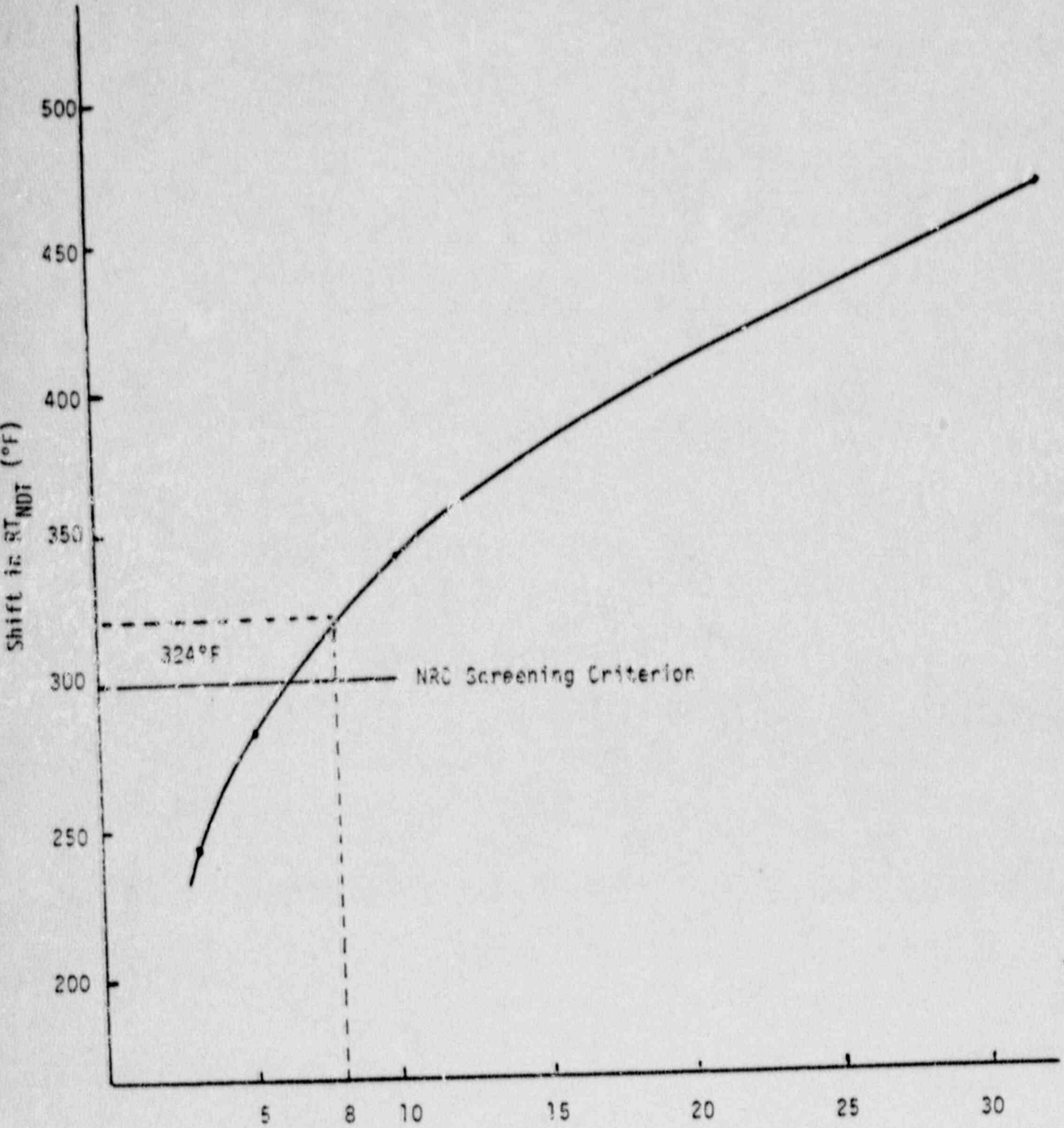
Very sincerely yours,

George C. Sih
Professor of Mechanics

GCS:bd

Enclosure

Data Reproduced from Table on Page 3 at Wall Location 1/4T, Report by E. B. Norris, "Reactor Vessel Material Surveillance Program for Turkey Point Unit No. 4: Analysis of Capsule T", Southwest Research Institute Technical Report No. 02-4221, June 1976.



REACTOR VESSEL MATERIAL SURVEILLANCE PROGRAM

CAPSULE S — TURKEY POINT UNIT NO. 3
CAPSULE S — TURKEY POINT UNIT NO. 4

FINAL REPORT
SwRI Project No. 02-5131
SwRI Project No. 02-5030

to

Florida Power & Light Company
P. O. Box 3100
Miami, Florida 33101

May 1979

SOUTHWEST RESEARCH INSTITUTE
SAN ANTONIO HOUSTON

EXHIBIT
NO. 12

reactor vessel.(21) The projected fast neutron exposures resulting from the analyses of the second surveillance capsule (S) from each unit are in good agreement with those reported earlier.(14,15) Also, since the S capsules did not contain specimens representing the controlling (weld metal) beltline material, there is no basis for revising the projected values of RT_{NDT} used to develop the current set of heatup and cooldown limit curves.

F. Capsule Removal Schedule

A third capsule is scheduled for removal from each reactor vessel after 10 calendar years of operation. Based on the past operating histories of the Turkey Point nuclear power plants, 10 calendar years of operation should correspond to approximately 7 EFPY of operation. It is recommended that Capsule V, a Type II capsule containing weld metal specimens, be removed from each vessel at that time. The projected fast neutron fluence for the V capsules after 7 EFPY is 1.3×10^{19} cm^{-2} (E > 1 MeV), approximately twice the fluence received by the T capsules.(14,15) The data obtained from the V capsules should provide the information necessary to revise the heatup and cooldown limitations for operation beyond 10 EFPY of operation.

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

April 22, 1985

Docket Nos. 50-250
and 50-251

RECEIVED
MAY 1 1985
Newman & Holtzinger

Mr. J. W. Williams, Jr., Vice President
Nuclear Energy Department
Florida Power and Light Company
Post Office Box 14000
Juno Beach, Florida 33408

Dear Mr. Williams:

The Commission has issued the enclosed Amendment No. 112 to Facility Operating License No. DPR-31 and Amendment No. 106 to Facility Operating License No. DPR-41 for the Turkey Point Plant Units Nos. 3 and 4, respectively. The amendments consist of changes to the Technical Specifications in response to your application transmitted by letters dated February 8, 1985 and March 6, 1985.

These amendments revise the Technical Specifications to provide consistency in identification of the surveillance specimen capsules in the Technical Specifications and the actual surveillance specimen capsules. The surveillance specimen examination schedule is also modified to provide better information in accordance with the current regulations. The proposed changes combine the existing Reactor Materials Surveillance Program into a single integrated program which conforms to the requirements of 10 CFR 50, Appendices G and H. We have discussed concerns and actions necessary regarding future core designs and in-cavity dosimetry in Section III of our Safety Evaluation provided in support of the amendments.

Section II.C of 10 CFR 50 Appendix H, which was revised on July 26, 1983, permits an integrated surveillance program provided it meets the criteria specified and is approved by the Director, Office of Nuclear Reactor Regulation. We have indicated in our Safety Evaluation that the integrated surveillance program for the Turkey Point Plant permitted by the enclosed amendments meet the criteria specified in 10 CFR 50, Appendix H II.C. The Director, Office of Nuclear Reactor Regulation, has approved the enclosed amendments which authorize an integrated surveillance program at the Turkey Point Plant in accordance with the requirements of 10 CFR 50, Appendix H II.C.

EXHIBIT
NO. 13

A copy of the related Safety Evaluation is enclosed. A Notice of Issuance will be included in the Commission's next regular monthly Federal Register notice.

Sincerely,

Daniel G. McDonald Jr.

Daniel G. McDonald, Jr., Project Manager
Operating Reactors Branch #1
Division of Licensing

Enclosures:
1. Amendment No. 112 to DPR-31
2. Amendment No. 106 to DPR-41
3. Safety Evaluation

cc: w/enclosures
See next page

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 112 TO FACILITY OPERATING LICENSE NO. DPR-31

AND AMENDMENT NO. 106 TO FACILITY OPERATING LICENSE NO. DPR-41

FLORIDA POWER AND LIGHT COMPANY

TURKEY POINT UNIT NOS. 3 AND 4

DOCKET NOS. 50-250 AND 50-251

I. Introduction

In a letter from J. W. Williams, Jr. to D. G. Eisenhut, dated February 8, 1985, Florida Power & Light Company requested that the Turkey Point Units No. 3 and 4 Technical Specifications be amended to combine the reactor vessel material surveillance program for these units into a single integrated surveillance program. Additional information concerning the proposed change was provided by the licensee in a letter from J. W. Williams, Jr. to S. A. Varga dated March 6, 1985.

A revised Appendix H, 10 CFR 50 was published in the Federal Register on May 27, 1983 and became effective on July 26, 1983. Section II.C of the revised Appendix H permits an integrated surveillance program provided it is approved by the Director, Office of Nuclear Reactor Regulation. This section of Appendix H identifies the criteria to be used in evaluating the integrated surveillance program. The criteria are:

1. There must be substantial advantages to be gained, such as reduced power outages or reduced personnel exposure to radiation, as a direct result of not requiring surveillance capsules in all reactors in the set.

2. The design and operating features of the reactors in the set must be sufficiently similar to permit accurate comparisons of the predicted amount of radiation damage as a function of total power output.

3. There must be an adequate dosimetry program for each reactor.

4. There must be a contingency plan to assure that the surveillance program for each reactor will not be jeopardized by operation at reduced power level or by an extended outage of another reactor from which data are expected.

5. No reduction in the requirements for number of materials to be irradiated, specimen type, or number of specimens per reactor is permitted, but the amount of testing may be reduced if the initial results agree with predictions.

6. There must be adequate arrangement for data sharing between plants.

II. Evaluation

Each unit at Turkey Point began commercial operation with 8 surveillance capsules in each reactor vessel. Ten capsules contained forging material and six capsules contained weld metal, forging, and heat affected zone (HAZ) materials. To date, two capsules containing forging material and two capsules containing weld metal, forging, and HAZ materials were irradiated, removed from the vessel, and tested. The test results from the surveillance material indicate that the weld metal will sustain the most irradiation damage. Since, based on the initial test, the weld metal is more susceptible to irradiation damage than the forging material, the licensee has proposed to retain the capsules with forging material as standby specimens in the reactor vessel and test only those capsules with weld metal, forging, and HAZ materials. Since fewer capsules will be withdrawn than originally anticipated, the radiation exposure (ALARA) to plant personnel should be reduced.

The licensee's FSAR Volume 2 indicates that the materials and designs for the core, thermal shield, core barrel and vessel are the same for each unit at Turkey Point. Since the neutron energy spectrum is a function of geometry, materials, and core loading, the relative neutron spectrum for both reactors should be equivalent for equivalent core loadings. The licensee indicates that fuel management and cycle lengths for both units have been similar. Thus neutron spectra profiles at the peak fluence locations should be equivalent.

The neutron fluence, which is used to predict radiation damage, is calculated using PDQT power distribution data, and computer codes SORREL and DOT 4.3. As built dimensions and individual material properties are incorporated into the DOT 4.3 models. Hence, using these codes, the licensee will be able to predict radiation damage as a function of power output for each unit.

Each vessel has both in-capsule and in-cavity dosimetry, which will be used to verify the neutron spectra and the codes that were used to predict neutron fluence as a function of power output. Since each plant has its own capsules and both plants are capable of independently predicting and monitoring radiation damage as a function of power output, the program will not be significantly jeopardized by operation at reduced power levels or by an extended outage of either plant.

Based on the intial test, the limiting material for each unit is weld material, which is identified as SA 1101. This material is in each capsule that will be irradiated and tested. Capsules that have been deleted from surveillance testing do not contain the limiting material and will be retained as standby specimens in the reactor vessel. Since the amount of limiting material in the surveillance program has not chnaged, the number of useful surveillance specimens available for testing has not changed.

Both units have common management and the surveillance program will be managed by their Nuclear Energy Department. Therefore, there should be adequate data sharing.

III. Findings

1. We have concluded based on the details in Section II of this Safety Evaluation, that the integrated surveillance program meets the evaluation criteria specified in 10 CFR 50, Appendix H II.C. If future core designs are significantly different than those documented by the licensee, the licensee must explain the effect that the changes have on neutron irradiation damage and the surveillance capsule withdrawal schedule.

2. In-cavity dosimetry testing should continue in order to reduce projected uncertainties in neutron fluence. If those test results provide an effective method of monitoring vessel neutron fluence, the in-cavity dosimetry should be incorporated into the integrated surveillance program.

IV. Environmental Consideration

These amendments involve changes in the installation or use of the facilities components located within the restricted areas as defined in 10 CFR 20 and in surveillance requirements. The staff has determined that these amendments involve no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that these amendments involve no significant hazards consideration and there has been no public comment on such finding. Accordingly, these amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR Sec 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of these amendments.

V. Conclusion

We have concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (2) such activities will be conducted in compliance with the Commission's regulations and the issuance of these amendments will not be inimical to the common defense and security or to the health and safety of the public.

Dated: April 22, 1985

Principal Contributors:

B. Elliot

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

February 27, 1985

Docket Nos. 50-250
and 50-251

Mr. J. W. Williams, Jr., Vice President
Nuclear Energy Department
Florida Power and Light Company
Post Office Box 14000
Juno Beach, Florida 33408

Dear Mr. Williams:

Reference: TAC Nos. 54428 and 55035

SUBJECT: NEAR TERM FLUX REDUCTION PROGRAM - TURKEY POINT PLANT UNITS 3 & 4

By letters dated March 1, 1984, April 2, 1984, June 4, 1984 and August 22, 1984, you provided the integral neutron source data we requested in our letters of November 17, 1983 and July 26, 1984. We have evaluated the data to verify the near term flux reduction resulting from your Pressurized Thermal Shock (PTS) program for the Turkey Point Plant.

The results of our initial Safety Evaluation (SE) are provided in Enclosure 1 to this letter. In reviewing your near term flux reduction program, we assessed the performance of the part-length burnable absorber assemblies designed explicitly for flux reduction to the pressure vessel circumferential welds and concluded that the flux reduction factor is 2.6. This conclusion was based on independent audit calculations performed by our technical consultants at Brookhaven National Laboratory.

However, our initial evaluation did not take into account the revised value of the required fast neutron fluence for Turkey Point Plant, Units 3 and 4, to reach the PTS screening criterion. The revised value is based on the details provided in our SE relating to Reactor Vessel Materials Data for the Turkey Point Vessels which was provided to you in our letter dated April 26, 1984.

The results of our supplemental SE, provided in Enclosure 2, indicates that the combination of the new fluence value and the present loading flux reduction will allow both plants to operate for 32 Effective Full Power Years (EFPY) without reaching the PTS screening criterion. The 32 EFPY is

EXHIBIT
NO. 14

equivalent to the 40 year licensed life considering a conservative capacity factor of 80%. This conclusion is based on the current low leakage loading factor. This completes our review of your near term flux reduction program.

Sincerely,

Steven A. Varga, Chief
Operating Reactors Branch #1
Division of Licensing

Enclosures:
As stated

cc w/enclosures:
See next page

J. W. Williams, Jr.
Florida Power and Light Company

Turkey Point Plants
Units 3 and 4

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TURKEY POINT UNITS 3 AND 4, EVALUATION OF THE FLUX REDUCTION FACTOR USING PART-LENGTH BURNABLE ABSORBER ASSEMBLIES TO MEET THE NRC PRESSURIZED THERMAL SHOCK CRITERIA

Introduction

The staff identified several plants in need of flux reduction in order for them to be able to operate for 32 Effective Full Power Years (EFPY) without violating the NRC Pressurized Thermal Shock (PTS) screening criteria. (1, 2). For Turkey Point - 3 and 4 the staff estimated (for the end of 1982) that the required flux reduction needed for either unit to operate for 40 calendar years (at a load factor of .8) was 4.5. Florida Power and Light (FP&L) the licensee has implemented a fluence reduction program consisting of low leakage fuel loading patterns coupled with part-length burnable absorbers, located so as to reduce the neutron flux to the pressure vessel circumferential weld from high importance core locations.

Based on power and exposure distributions supplied by FP&L (3-7), the Core Performance Branch performed an evaluation of the fluxes (and fluences) associated with the first nine cycles of operation of Unit 4 and the first 10 cycles of operation of Unit 3. The review and evaluation included independent audit calculations carried out by staff consultants at BNL.

Evaluation

Fast neutron flux ($E > 1.0$ MeV) calculations at the inner surface of the Pressure Vessel (PV) on the lower core belt circumferential weld were based on the flux synthesis methodology (8).

This approach consists of the following steps:

a. Determine the contributions to the flux above 1.0 MeV near 0° (the peak azimuthal flux location) on the inner surface of the PV from individual assemblies in the reactor core based on calculations in (r,θ) geometry.

b. Determine the contributions to the fast flux at the lower-to-intermediate shell circumferential weld from discrete 12 in. high axial segments for the two outermost rows of assemblies based on calcualtions in (r,z) geometry.

c. Combine the results from (1) and (2) with the three-dimensional core power (neutron source) distributions to obtain the desired flux and fluence values.

The same approach was also used for H. B. Robinson and the (r,θ) geometrical results have been used here as well. These results were generated with the DOT-3.5 (9) discrete ordinates transport code in the fixed-source mode with an S_8-P_3 angular approximation. Region dependent, 16 neutron group cross sections were based on the DLC-37/EPR (ENDF/B-IV) library (10). HBR-2 has virtually identical core/internals/vessel dimensions and materials to those of the Turkey Point units; therefore, the only modification to the HBR-2 results was a slight increase in the flux values to account for the higher temperature of the bypass water for the Turkey Point units. The results of these calculations provided the flux above 1.0 MeV at the inner surface of the PV near the core major axis due to unit sources located in assemblies 6, 7, 8, 13, 14, 15, 19, 20 and 24, Figure 1.

Calculations were also performed in the (r,z) geometry with the reactor axial configuration as shown in Figure 2. This configuration was modelled with 91 axial and 78 radial intervals with the DOT-4.3 (11) discrete ordinates transport code.

The 16-group, P_3 cross sections were the same as those used for the (r,θ) calculations. Note that a single set of cross sections was used for the core, i.e. axially zoned burnable absorbers were not accounted for. Fixed source calculations were performed in the adjoint mode with an S_8 symmetric quadrature. The fixed source was located at the inner surface of the vessel at the elevation of the limiting circumferential weld (Figure 2) and the importance of 12 in. high axial segments in the first and second outermost rows of assemblies to the fast flux at the weld were determined. Finally the (r,θ) and (r,z) geometry results were combined with the core power distributions to obtain the flux above 1.0 MeV at the limiting circumferential weld near the core major axis. A further refinement was included i.e. an exposure correction based on the analysis of Reference 12.

Power and exposure distribution data were provided by FP&L for the determination of the sources to be used in the evaluation of present and projected EOL fluences. While the information that was provided was relatively complete for Unit-3, not all the necessary assembly exposure data were available for all cycles of Unit 4. Consequently, reasonable estimates were made for the average exposure associated with the peripheral assemblies for cycles for which this data had not been provided. The only other area where approximations for the source were made for both units was related to the axial power distributions since data were not provided for all assemblies required in the flux synthesis scheme.

Results for the fast flux at the limiting circumferential weld near the core major axis are presented in Table 1 for Turkey Point Units 3 and 4. Results are for Cycles 1-7 (based on single exposure weighted source and exposure distributions) and for Cycle 8, and 9, and for Unit-4, Cycle 10, explicitly. Two sets of results are given for each cycle, one assuming a uniform nominal exposure of 6,000 MWD/MTU for all assemblies, and one where the assembly-wise neutron sources were corrected for the specific exposures associated with each assembly.

The results in Table 1 account for the neglect of pin-wise source distribution effects on the (r,θ) DOT calculation by an approximate factor based on a generic study of this effect (12). The percent increase in the fast flux due to the exposure correction, and fast flux reduction factors for cycles greater than Cycle 7, relative to the results for the averaged Cycle 1-7, are also given.

The associated estimates for the accumulated fluence after each cycle and at EOL (assumed to be 32 (EFPY) are given in Table 2. These values are based on the exposure corrected fast flux values of Table 1. The results indicate that a significant reduction in the fast flux (~ 62%) can be achieved at the critical weld by a combination of an "extreme" low leakage fuel loading pattern coupled with appropriately located part-length absorbers (in assemblies 8 and 15 of Figure 1). The reduction in projected EOL fluence, however, is less (~50%) relative to the value obtained by assuming that the averag[illegible] Cycle 1-7 power distribution is applicable through life.

A reduction of the fast flux by 62% is equivalent to a factor [illegible]3. If the flux reduction which was implemented for Cycle 8 in Unit 3 a[illegible] 9 in Unit 4, were maintained both units would reach the screening criterion in 1989. (assuming an 80% load factor) (13). According to the August 2, 1983 licensee presentation to the staff, progressively higher flux reduction factors were planned for both units. A flux reduction factor of 2.2 will extend the date to 1994, while a factor of 3.3 will extend it to 2007. However, our estimate of the flux reduction based on the FP&L data is 2.63 which corresponds to 1999.

Summary and Conclusion

An audit calculation was performed by BNL on behalf of the staff to evaluate the performance of the proposed part-length burnable absorber assemblies with respect to fast neutron flux reduction to the pressure vessel. The methodolgy employed by BNL was based on three dimensional flux synthesis. Based on data supplied by Florida Power and Light it was estimated that the maximum flux reduction was by a factor of 2.63. Assuming an 80% load factor this would enable both units to meet the PTS scre[illegible] criteria until 1999.

Principal Contributor:

L. Lois

WCAP-11138 WESTINGHOUSE CLASS 3

CUSTOMER DESIGNATED DISTRIBUTION

REACTOR CAVITY NEUTRON MEASUREMENT PROGRAM
FOR
FLORIDA POWER AND LIGHT COMPANY
TURKEY POINT UNIT 3

S. L. Anderson

A. H. Fero

E. P. Lippincott

April 1986

Work performed under Shop Order No. FJVP-450 and FIUP-450

APPROVED: F. L. Lau

F. L. Lau, Manager
Radiation and Systems Analysis

Prepared by Westinghouse for the Florida Power and Light Company

WESTINGHOUSE ELECTRIC CORPORATION
Nuclear Energy Systems
P.O. Box 355
Pittsburgh, Pennsylvania 15230

3618e:1d/050586

SECTION 1

PROGRAM OVERVIEW

1-1. INTRODUCTION

The Reactor Cavity Neutron Measurement Program at Turkey Point Unit 3 is designed to provide a mechanism for the long term monitoring of the neutron exposure of those portions of the reactor vessel and vessel support structure which may experience radiation induced increases in reference nil ductility transition temperature (RT_{NDT}) over the nuclear power plant lifetime. When used in conjunction with dosimetry from internal surveillance capsules and with the results of neutron transport calculations, the reactor cavity dosimetry allows the projection of embrittlement gradients through the reactor vessel wall with a minimum uncertainty. Minimizing the uncertainty in the neutron exposure projections will, in turn, help to assure that the reactor can be operated in the least restrictive mode possible with respect to

1. 10CFR50 Appendix G pressure/temperature limit curves for normal heatup and cooldown of the reactor coolant system.

2. Emergency Response Guideline (ERG) pressure/temperature limit curves.

3. Pressurized Thermal Shock (PTS) RT_{NDT} screening criteria.

In addition, an accurate measure of the neutron exposure of the reactor vessel and support structure can provide a sound basis for requalification should operation of the plant beyond the current design and/or licensed lifetime prove to be desirable.

1-2. BACKGROUND

Over the lifetime of a nuclear power plant, changing fuel management schemes can result in significant changes in both the magnitude and distribution of neutron flux and, hence, neutron fluence throughout the reactor vessel beltline region. In order to accurately assess the long-term effects of neutron irradiation on reactor vessel materials properties, these changes in radiation level must be well known.

Each operating reactor currently has a reactor vessel surveillance program usually consisting of from six to eight surveillance capsules located between the core and the reactor vessel in the downcomer region near the reactor vessel wall. The neutron dosimeters contained in these surveillance capsules provide measurement capability at a single location within the reactor geometry. By themselves they cannot provide the gradient information that is required to evaluate the impact of fuel management schemes (such as the incorporation of low leakage loading patterns) which may result in radical changes in neutron flux distributions from cycle to cycle.

Additional information can be obtained by the use of supplementary passive neutron dosimeters installed in the reactor cavity annulus between the reactor vessel wall and the primary shield.

This dosimetry package provides spectral coverage sufficient to allow the determination of fast neutron exposure parameters in terms of both neutron fluence (E > 1.0 MeV) and iron displacements per atom (dpa). The results of this program will establish the azimuthal and axial gradients of fast neutron flux and dpa over the beltline region of the reactor vessel, and will provide a verification of the ability of neutron transport analyses to predict through-wall embrittlement gradients.

1-3. TECHNICAL DESCRIPTION

To achieve the goals of the Reactor Cavity Neutron Measurement Program two types of measurements are made. Comprehensive sensor sets including radiometric monitors (RM) and solid state track recorders (SSTR) are employed at discrete locations within the reactor cavity to characterize the neutron energy spectrum variations axially and azimuthally over the beltline region of

RELOAD SAFETY EVALUATION

TURKEY POINT PLANT UNIT 3, CYCLE 10

February 1985

Edited by

M. J. Weber

1.0 INTRODUCTION AND SUMMARY

1.1 Introduction

This report presents an evaluation for Turkey Point 3, Cycle 10, which demonstrates that the core reload will not adversely affect the safety of the plant. This evaluation was accomplished utilizing the methodology described in WCAP-9273, "Westinghouse Reload Safety Evaluation Methodology"(1).

Turkey Point Unit 3 is operating in Cycle 9 with 56 Westinghouse optimized fuel assemblies and 101 Westinghouse 15x15 low parasitic (LOPAR) fuel assemblies. For Cycle 10 (expected startup June 19, 1985) and subsequent cycles, it is planned to refuel the Turkey Point Unit 3 core with Westinghouse 15x15 optimized fuel assembly (OFA) regions. In a licensing submittal (2) to the NRC, approval was requested and later approved for the transition from LOPAR fuel to OFA and associated proposed changes to the Turkey Point Units 3 and 4 Technical Specifications. The licensing submittal justified the compatibility of Optimized Fuel Assemblies (OFAs) with LOPAR fuel assemblies in a mixed-fuel core as well as a full OFA core. The licensing submittal contained mechanical, nuclear, thermal-hydraulic, and accident evaluations which are also applicable to the Cycle 10 safety evaluation. Approval of the license application for the OFA transition was granted by the NRC in a SER(3) dated December 9, 1983.

All of the accidents comprising the licensing bases(2,7) which could potentially be affected by the fuel reload have been reviewed for the Cycle 10 design described herein. The results of new analyses are

RSE-004A

RELOAD SAFETY EVALUATION

TURKEY POINT PLANT UNIT 4, CYCLE 10

REDESIGN

[illegible]

Westinghouse Nuclear Energy Systems

RSE - 004A

RELOAD SAFETY EVALUATION

TURKEY POINT PLANT UNIT 4, CYCLE 1[illegible]

REDESIGN

Westinghouse Nuclear Energy Systems

EXHIBIT

NO. 17

AEOD/C401

LOW TEMPERATURE OVERPRESSURE EVENTS AT TURKEY POINT UNIT 4

Case Study Report
Reactor Operations Analysis Branch

Office for Analysis and Evaluation
of Operational Data

March 1984

Prepared by: Wayne D. Lanning

1.0 INTRODUCTION

Before 1979, 30 reported incidents occurred in pressurized water reactors (PWRs) where the pressure/temperature limits contained in the technical specifications for the reactor coolant system were exceeded. Most of these events occurred during reactor startup or shutdown when the reactor coolant system was in a water solid condition, i.e., no steam or gas space in the pressurizer. Overpressure events primarily resulted from the loss of letdown flow with continued charging flow, inadvertent safety injection, or a heatup transient caused by starting a reactor coolant pump with the secondary coolant system temperature higher than the primary temperature. These events were caused by either equipment malfunction or operator error.

Low temperature overpressurization (LTOP) was designated a generic issue because of the possibility of a vessel failing by the brittle fracture mechanism. This failure mode may be a consequence of a pressure transient after the vessel material toughness has been reduced due to irradiation effects (i.e., increase in nil-ductility transition temperature) while a critical size flaw exists in the vessel wall. NRC resolved the generic issue in 1979* by recommending that PWR licensees implement procedures to reduce the potential for overpressure events and install equipment modifications to mitigate such events.

Since that time, ten pressure transients have been reported. The two events at Turkey Point Unit 4 on November 28 and 29, 1981 exceeded the technical specification limit (415 psig below 355°F) by about 700 and 325 psi, respectively. The two events were designated Abnormal Occurrences by the NRC (Ref. 1). The other eight reported events were mitigated by the overpressure protection system. These two overpressure events and a significant number of events at other PWRs involving inoperable trains of the overpressure protection system prompted AEOD to initiate an evaluation of operational events with the focus primarily on Turkey Point.

The overpressure protection system and the overpressure events at Turkey Point Unit 4 are described in Sections 2 and 3. Section 4 contains the analyses and evaluation of the two events, including utility management's reaction to the events. Section 5 reviews the operational experience related to inoperable trains of the overpressure protection system at other PWRs. Section 6 evaluates the adequacy of existing LTOP technical specifications. Section 7 discusses the need for operating in a water solid condition. Section 8 lists the findings and conclusions, and Section 9 contains the AEOD recommendations based on this case study.

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*NUREG-0224 entitled, "Reactor Vessel Pressure Transient Protection for Pressurized Water Reactors," was published in September 1978 documenting the completion of the generic activity. LTOP mitigating systems were installed in most plants beginning in 1979.

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In a separate licensing submittal[4] to the NRC, approval was requested to increase the maximum $F_{\Delta H}$ limit to 1.62 at normal operating conditions as part of a vessel flux reduction program[5] to partially resolve the pressurized thermal shock concerns. The report contains nuclear, thermal-hydraulic, and accident evaluations which are applicable to the Cycle 10 safety evaluation. Approval of the license application[4] for the increase in the $F^N_{\Delta H}$ limit was granted by the NRC in a SER(6) dated December 23, 1983.

RELOAD SAFETY EVALUATION

TURKEY POINT PLANT UNIT 3, CYCLE 11

REVISION 1

April 1987

Westinghouse Nuclear Energy Systems

EXHIBIT

NO. 18

RELOAD SAFETY EVALUATION

TURKEY POINT PLANT UNIT 3, CYCLE 11

REVISION 1

April 1987

Westinghouse Nuclear Energy Systems

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Turkey Point Unit 3 is operating in Cycle 10 with 112 Westinghouse optimized fuel assemblies and 45 Westinghouse 15x15 low parasitic (LOPAR) fuel assemblies. For Cycle 11 (expected startup mid-May, 1987) and subsequent cycles, it is planned to refuel the Turkey Point Unit 3 core with Westinghouse 15x15 optimized fuel assembly (OFA) regions. In a licensing submittal [2] to the NRC, approval was requested and later received for the transition from LOPAR fuel to OFA and the associated proposed changes to the Turkey Point Units 3 and 4 Technical Specifications. The licensing submittal justified the compatibility of Optimized Fuel Assemblies (OFAs) with LOPAR fuel assemblies in a mixed-fuel core as well as a full OFA core. The licensing submittal contained mechanical, nuclear, thermal-hydraulic, and accident evaluations which are also applicable to the Cycle 11 safety evaluation. Approval of the license application for the OFA transition was granted by the NRC in a SER[3] dated December 9, 1983.

A significant number of Integral Fuel Burnable Absorber (IFBA) rods are being used for the first time in Turkey Point Unit 3* as part of the Region 13C and 13D fuel assemblies. These rods are described in Section 2.1. A more detailed description and evaluation of IFBAs for 14x14, 15x15 and 17x17 fuel arrays are given in References 4 and 5. The NRC has approved the use of IFBAs for Westinghouse fuel rods in 15x15 fuel assemblies[6].

*Turkey Point Unit 3 did have demonstration IFBA rods in Cycles 8 and 9.

RSE - 005A

RELOAD SAFETY EVALUATION
TURKEY POINT PLANT
UNIT 4, CYCLE 11
REVISION 1

April 1986

Edited by:

J. S. Baker
J. Skaritka

Approved: Edward A Dzenis
E. A. Dzenis, Manager
Core Operations
Nuclear Fuel Division

EXHIBIT
NO. 19

4424L 6-850422

1.0 INTRODUCTION AND SUMMARY

1.1 INTRODUCTION

This report presents an evaluation for Turkey Point Unit 4, Cycle 11, which demonstrates that the core reload will not adversely affect the safety of the plant. This evaluation was accomplished utilizing the methodology described in WCAP-9273, "Westinghouse Reload Safety Evaluation Methodology"[1].

Turkey Point Unit 4 operated during Cycle 10 with 117 Westinghouse 15x15 low parasitic (LOPAR) fuel assemblies and 40 Westinghouse 15x15 optimized fuel assemblies (OFA). For Cycle 11 (expected startup May 1986) and subsequent cycles, it is planned to refuel the Turkey Point Unit 4 core with primarily Westinghouse 15x15 optimized fuel assembly (OFA) regions. In a licensing submittal[2] to the NRC, approval was requested for the transition from LOPAR fuel to OFA and associated proposed changes to the Turkey Point Units 3 and 4 Technical Specifications. The licensing submittal justifies the compatibility of OFAs and LOPAR fuel assemblies in a mixed-fuel core as well as a full OFA core. The licensing submittal contains mechanical, nuclear, thermal-hydraulic, and accident evaluations which are applicable to the Cycle 11 safety evaluation. Approval of the license application[2] for the OFA transition was granted by the NRC in a SER[3] dated December 9, 1983.

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Steel Hector & Davis
Miami, Florida

John T. Butler
(305) 577-2939

October 13, 1989

Joette Lorion
Center for Nuclear Responsibility
5901 S.W. 74th Street
Suite #304
South Miami, Florida 33143

Re: Florida Power & Light Company (Turkey Point Plant, Units 3 and 4), Docket Nos. 50-250-OLA-4 and 50-251-OLA-4 (P/T Limits)

Dear Joette:

I am enclosing copies of the safety evaluations for the Unit 4, Cycles 10 and 11 fuel reloads. Together with the safety evaluations previously delivered to you, you should now have the safety evaluations for Unit 3, Cycles 9, 10 and 11, and for Unit 4, Cycles 10, 11 and 12. These represent the evaluations for cycles that covered the period beginning in 1985 and extending to the present.

You also asked me for the capacity factors for years prior to 1985. I believe the following is responsive to your request (1974 was the first year for which the information was available to me):

	Unit 3	Unit 4
1974	62.1	74.1
1975	75.0	68.4
1976	73.8	64.5
1977	76.6	62.8
1978	77.1	64.9
1979	49.3	65.9
1980	77.3	67.9
1981	16.1	78.5
1982	66.5	67.9
1983	75.0	51.7
1984	81.8	52.6

My records reflect that you now have all the information you requested. Please contact me if this is not

EXHIBIT
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your understanding as well. I apologize for the earlier confusion and hope that, by providing the missing information to you within a day of your request, I have avoided any serious inconvenience on your part.

Sincerely,

John T. Butler

Enclosures

cc: Steven P. Frantz, co-counsel for
Florida Power & Light Company

DO NOT CIRCULATE

FPL

FLORIDA POWER & LIGHT COMPANY

22

February 8, 1985
L-85-66

Office of Nuclear Reactor Regulation
Attention: Mr. Darrell G. Eisenhut, Director
Division of Licensing
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Dear Mr. Eisenhut:

Re: Turkey Point Units 3 and 4
Docket Nos. 50-250 and 50-251
Proposed License Amendment
Reactor Plant Surveillance Material Program

In accordance with 10 CFR 50.90, Florida Power & Light Company submits herewith three signed originals and forty copies of a request to amend Appendix A of Facility Operating Licenses DPR-31 and 41.

This amendment is proposed to combine the reactor materials surveillance program at Unit 3 and 4 into a single integrated program which conforms to the requirements of 10 CFR 50 Appendices G and H.

The proposed amendment is described below and shown on the accompanying Technical Specification pages.

Table 4.2-1 and Page 4.20-1

The Irradiation Specimen Schedule (item 7-2) on Table 4.2-1, is deleted and a revised version to reflect the proposed integrated program is added to Page 4.20-1.

Pages B 3.1-3, B 4.2-12, B 4.2-13, B 4.20-1 and delete B 4.2-14

The bases associated with the above changes are revised.

The proposed amendment has been reviewed by the Turkey Point Plant Nuclear Safety Committee and the Florida Power & Light Company Nuclear Review Board.

FPL requests issuance of this proposed amendment before the beginning of the Spring 1985 Unit 3 refueling outage (currently scheduled to begin 3-30-85) in order to allow proper implementation of the single integrated program.

In accordance with 10 CFR 50.91(b)(I), a copy of the proposed amendment is being forwarded to the State Designee for the State of Florida.

Rec'd w/check $150.00 #6332

A001

3/7

8502200344 850208
PDR ADOCK 05000250
P PDR

EXHIBIT
NO. 21

ccordance with 10 CFR 170.12(c), a check for $150 is attached.

attached is an evaluation of the proposed action in light of the three andards contained in 10 CFR 50.92 (No Significant Hazards).

ry truly yours,

. W. Williams, Jr.
roup Vice President
uclear Energy

W/SAV/js

ttachment

cc: J. Nelson Grace, Region II
Harold F. Reis, Esquire
Lyle Jerrett, Ph.D., Director
Office of Radiation Control
Dept. of Health & Rehabilitative Services
1323 Winewood Boulevard
Tallahassee, FL 32301

PNS-LI-85-058-1

REACTOR MATERIAL SURVEILLANCE PROGRAM

Each Type I capsule contains 28 Charpy V-notch specimens, ten Charpy specimens machined from each of the two shell forgings. The remaining eight Charpy specimens are machined from correlated monitor material. In addition, each Type I capsule contains four tensile specimens (two specimens from each of the two shell forgings) and six WOL specimens (three specimens from each of the two shell forgings). Dosimeters of copper, nickel, aluminum-cobalt, and cadmium-shielded aluminum-cobalt wire are secured in holes drilled in spacers at the top, middle, and bottom of each Type I capsule.

Each Type II capsule contains 32 Charpy V-notch specimens: eight specimens machined from one of the shell forgings, eight specimens of weld metal and eight specimens of HAZ metal, the remaining eight specimens are correlation monitors. In addition, each Type II capsule contains four tensile specimens and four WOL specimens: two tensile specimens and two WOL specimens from one of the shell forgings and the weld metal. Each Type II capsule contains a dosimeter block at the center of the capsule. Two cadmium-oxide-shielded capsules, containing the two isotopes uranium-238 and neptunium-237, are contained in the dosimeter block. The double containment afforded by the dosimeter assembly prevents loss and contamination by the neptunium-237 and uranium-238 and their activation products. Each dosimeter block contains approximately 20 milligrams of neptunium-237 and 13 milligrams of uranium-238 contained in a 3/8-inch-OD sealed brass tube. Each tube is placed in a 1/2-inch-diameter hole in the dosimeter block (one neptunium-237 and one uranium-238 tube per block), and the space around the tube is filled with cadmium oxide. After placement of this material, each hole is blocked with two 1/16-inch aluminum spacer discs and an outer 1/8-inch-steel cover disc, which is welded in place. Dosimeters of copper, nickel, aluminum-cobalt, and cadmium-shielded aluminum-cobalt are also secured in holes drilled in spacers located at the top, middle, and bottom of each Type II capsule.

Capsule Type	Capsule Identification
I	S
II	V
II	T
I	U
II	X
I	W
I	Y
I	Z

This program combines the Reactor Materials Surveillance Program into a single integrated program which conforms to the requirements of 10CFR50 Appendices G and H.

TURKEY POINT UNITS 3 AND 4

REACTOR SURVEILLANCE MATERIAL PROGRAM

PROPOSED CHANGE TO PLANT TECHNICAL SPECIFICATIONS

Appendix H requires reactors constructed of ferritic materials have their beltline regions monitored by a surveillance program complying with ASTM E185. Appendix G defines beltline materials as shell material including welds and heat affected zones, plates or forgings, that directly surround the effective height of the fuel element assemblies.

The existing Turkey Point 3 and 4 surveillance programs contain two types of surveillance capsules: 5 Type I capsules contain forging samples only; 3 Type II capsules contain forging, weld, and haz samples.

The first Type II capsule removed has defined the most limiting material in the reactor as the girth welds based on fracture toughness requirements.

Attachment 1 is an excerpt from the PTP surveillance program. Attachment 2 shows the number and identification system of Type I and II capsules in each of the Turkey Point Vessels. As can be seen, there are only two Type II capsules remaining in each vessel. Attachment 3 shows the capsule locations.

To obtain the most meaningful results from the existing program and to update the program to current Appendix H requirements, FPL proposes to remove only Type II surveillance capsules for the remainder of plant life. This requires that 3 capsules be available for removal through the end of life. Since there are only 2 capsules available for each unit, we propose to integrate the surveillance programs as permitted by Appendix H, II, C.

The requirements of 10 CFR 50 Appendix H, II, C, are:

1) Degree of Commonality

a) Design

PTP 3 and 4 are identical in design, share identical Plant Technical Specifications and have had identical major modifications such as steam generator replacement and TMI backfit modifications. The reactor vessels were fabricated the same way by the same supplier utilizing the same materials.

b) Materials

All reactor materials of fabrication are identical using SA508 C12, SA 302 grade B ferritic steels. The intermediate to lower shell girth welds were made by automatic submerged arc welding using Linde 80 flux (lot No. 8445) and Page copper coated weld wire (heat No. 71249) and identified as SA1101. Since this weld is the material with the highest predicted RT_{NDT} and is identical for both units, it is our opinion that these units are particularly well suited to an integrated surveillance program.

c) Predicted Severity of Irradiation

Both reactor vessels are expected to experience an end of life fluence of a maximum of 1.8x1019 n/cm2 (E≥ 1 mev) and have operated using similar fuel loading since startup.

The Turkey Point inner wall vessel fluence predictions are 8.19x1018 in April 1985 for Units 3 and 8.4x1018 in October 1985 for Unit 4. The difference is due to different service EFPY.

We have installed excore dosimetry around both Turkey Point vessels to benchmark individual cycle fluence, thereby reducing our dependence on incore surveillance capsule foil dosimetry.

2) Data Sharing Between Plants

Both units have common management, and the surveillance programs are managed by the Codes and Inspections section of the Nuclear Energy Department staff.

3) Contingency Plan in the Event of Reduced Power Operations or Extended Outage

Both plants have capsules.

4) Substantial Advantages To Be Gained

The main advantage is obtaining the best data available from each capsule removal. Additional advantage will be realized from fewer capsule removals and both plants operating to identical heat up and cool down pressure temperature curves.

ALARA benefits also exist since fewer capsules will be removed over plant life.

Applicant Correspondence File

TO BE FILED IN

FPL

FLORIDA POWER & LIGHT COMPANY

April 11, 1977
L-77-113

RECEIVED APR 14 1977

Office of Nuclear Reactor Regulation
Attention: Mr. George Lear, Chief
Operating Reactors Branch #3
Division of Operating Reactors
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Dear Mr. Lear:

Re: Turkey Point Unit 4
Docket No. 50-251
Fracture Toughness Requirements

On April 7, 1977, a meeting was held with members of your staff to discuss the status of the Turkey Point Unit 4 reactor vessel with respect to the fracture toughness requirements of Section V.B of Appendix G to 10 CFR 50. At that meeting, we showed that the weld metal surveillance data for the Turkey Point Unit 3 reactor vessel represent not only the core midplane circumferential welds in Unit 3, but in Unit 4 as well. Data supporting this conclusion are attached.

The data show that the weldment samples from a Unit 3 surveillance capsule "T" and from both the Unit 3 and Unit 4 reactor vessels were made from the same combination of filler wire heat number and welding flux lot number. However, the weldment samples from a Unit 4 surveillance capsule "T", although containing the same filler wire heat number, used a different welding flux lot number. Therefore, the Unit 3 capsule "T" sample is more representative of the Unit 4 reactor vessel.

Irradiation data from the Unit 3 capsule was submitted to the NRC on October 19, 1976 (L-75-363). The data exhibited a shelf energy of 58 ft-lbs at a fluence of 5.7×10^{18} nvt. Accordingly, the mid-plane circumferential vessel weld in Unit 4 can be expected to maintain a shelf energy level in excess of 50 ft-lbs at the 1/4 T location until at least June 1980 at which time this location will have received a fluence of 5.7×10^{18} nvt.

EXHIBIT
NO. 22

77050306

PEOPLE ... SERVING PEOPLE

In the October 19 letter, we also stated that additional re... ts were being prepared by our NSSS vendor to complete summari... the fatigue, accident, and fracture analyses for Units 3 an... . We expect to receive these additional reports in draft form ... about one week, and should be able to forward them on to yo... office in approximately 6 to 8 weeks.

The evaluation discussed above supports the conclusion we presented at the April 7 meeting that an Appendix G inservi... inspection of the Unit 4 reactor vessel belt-line area need not be conducted until after June 1980.

Very truly yours,

Robert E. Uhrig
Vice President

REU/MAS/cpc

Attachment

cc: Mr. Norman C. Moseley, Region II
Robert Lowenstein, Esquire

AEOD/C401

LOW TEMPERATURE OVERPRESSURE EVENTS AT TURKEY POINT UNIT 4

Case Study Report
Reactor Operations Analysis Branch

Office for Analysis and Evaluation
of Operational Data

March 1984

Prepared by: Wayne D. Lanning

1.0 INTRODUCTION

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RELOAD SAFETY EVALUATION

TURKEY POINT PLANT UNIT 3, CYCLE 11

REVISION 1

April 1987

Westinghouse Nuclear Energy Systems

EXHIBIT

NO. 18

RELOAD SAFETY EVALUATION

TURKEY POINT PLANT UNIT 3, CYCLE 11

REVISION 1

April 1987

Westinghouse Nuclear Energy Systems

EXHIBIT

NO. 18

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*Turkey Point Unit 3 did have demonstration IFBA rods in Cycles 8 and 9.

RSE-005A

RELOAD SAFETY EVALUATION
TURKEY POINT PLANT
UNIT 4, CYCLE 11
REVISION 1

April 1985

Edited by:

J. S. Baker

J. Skaritka

Approved: [signature]

E. A. Dzenis, Manager

Core Operations

Nuclear Fuel Division

EXHIBIT NO. 19

4424, 6-850422

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1.1 INTRODUCTION

This report presents an evaluation for Turkey Point Unit 4, Cycle 11, which demonstrates that the core reload will not adversely affect the safety of the plant. This evaluation was accomplished utilizing the methodology described in WCAP-9273, "Westinghouse Reload Safety Evaluation Methodology"[1].

Turkey Point Unit 4 operated during Cycle 10 with 117 Westinghouse 15x15 low parasitic (LOPAR) fuel assemblies and 40 Westinghouse 15x15 optimized fuel assemblies (OFA). For Cycle 11 (expected startup May 1986) and subsequent cycles, it is planned to refuel the Turkey Point Unit 4 core with primarily Westinghouse 15x15 optimized fuel assembly (OFA) regions. In a licensing submittal[2] to the NRC, approval was requested for the transition from LOPAR fuel to OFA and associated proposed changes to the Turkey Point Units 3 and 4 Technical Specifications. The licensing submittal justifies the compatibility of OFAs and LOPAR fuel assemblies in a mixed-fuel core as well as a full OFA core. The licensing submittal contains mechanical, nuclear, thermal-hydraulic, and accident evaluations which are applicable to the Cycle 11 safety evaluation. Approval of the license application[2] for the OFA transition was granted by the NRC in a SER[3] dated December 9, 1983.

In a separate licensing submittal[4] to the NRC, approval was requested to increase the maximum $F_{\Delta H}$ limit to 1.62 at normal operating conditions as part of a vessel flux reduction program[5] to partially resolve the pressurized thermal shock concerns. The report contains nuclear, thermal-hydraulic, and accident evaluations which are applicable to the Cycle 11 safety evaluation. Approval of the license application[4] for the increase in the $F^N_{\Delta H}$ limit was granted by the NRC in a SER[6] dated December 23, 1983.

Steel Hector & Davis
Miami, Florida

John T. Butler
(305) 577-2939

October 13, 1989

Joette Lorion
Center for Nuclear Responsibility
5901 S.W. 74th Street
Suite #304
South Miami, Florida 33143

Re: Florida Power & Light Company (Turkey Point Plant, Units 3 and 4), Docket Nos. 50-250-OLA-4 and 50-251-OLA-4 (P/T Limits)

Dear Joette:

I am enclosing copies of the safety evaluations for the Unit 4, Cycles 10 and 11 fuel reloads. Together with the safety evaluations previously delivered to you, you should now have the safety evaluations for Unit 3, Cycles 9, 10 and 11, and for Unit 4, Cycles 10, 11 and 12. These represent the evaluations for cycles that covered the period beginning in 1985 and extending to the present.

You also asked me for the capacity factors for years prior to 1985. I believe the following is responsive to your request (1974 was the first year for which the information was available to me):

	Unit 3	Unit 4
1974	62.1	74.1
1975	75.0	68.4
1976	73.8	64.5
1977	76.6	62.8
1978	77.1	64.9
1979	49.3	65.9
1980	77.3	67.9
1981	16.1	78.5
1982	66.5	67.9
1983	75.0	51.7
1984	81.5	52.6

My records reflect that you now have all the information you requested. Please contact me if this is not

EXHIBIT
NO. 20

Miami Office
4000 Southeast Financial Center
Miami, FL 33131-2398
(305) 577-2800
Fax (305) 358-1418

1200 Northbridge Centre 1
West Palm Beach, FL 33401-4307
(305) 650-7200
Fax (305) 655-1509

440 Royal Palm Way
Palm Beach, FL 33480
(305) 650-7200

1200 Corporate Place
1200 North Federal Highway
Boca Raton, FL 33432
(305) 394-5000
Fax (305) 394-4856

201 South Monroe
Tallahassee, FL 32301-1848
(904) 222-4194
Fax (904) 222-8410

your understanding as well. I apologize for the earlier confusion and hope that, by providing the missing information to you within a day of your request, I have avoided any serious inconvenience on your part.

Sincerely,

JTB

John T. Butler

Enclosures

cc: Steven P. Frantz, co-counsel for
Florida Power & Light Company

DO NOT CIRCULATE

FPL

FLORIDA POWER & LIGHT COMPANY

22

February 8, 1985
L-85-66

Office of Nuclear Reactor Regulation
Attention: Mr. Darrell G. Eisenhut, Director
Division of Licensing
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Dear Mr. Eisenhut:

Re: Turkey Point Units 3 and 4
Docket Nos. 50-250 and 50-251
Proposed License Amendment
Reactor Plant Surveillance Material Program

In accordance with 10 CFR 50.90, Florida Power & Light Company submits herewith three signed originals and forty copies of a request to amend Appendix A of Facility Operating Licenses DPR-31 and 41.

This amendment is proposed to combine the reactor materials surveillance program at Unit 3 and 4 into a single integrated program which conforms to the requirements of 10 CFR 50 Appendices G and H.

The proposed amendment is described below and shown on the accompanying Technical Specification pages.

Table 4.2-1 and Page 4.20-1

The Irradiation Specimen Schedule (item 7-2) on Table 4.2-1, is deleted and a revised version to reflect the proposed integrated program is added to Page 4.20-1.

Pages B 3.1-3, B 4.2-12, B 4.2-13, B 4.20-1 and delete B 4.2-14

The bases associated with the above changes are revised.

The proposed amendment has been reviewed by the Turkey Point Plant Nuclear Safety Committee and the Florida Power & Light Company Nuclear Review Board.

FPL requests issuance of this proposed amendment before the beginning of the Spring 1985 Unit 3 refueling outage (currently scheduled to begin 3-30-85) in order to allow proper implementation of the single integrated program.

In accordance with 10 CFR 50.91(b)(I), a copy of the proposed amendment is being forwarded to the State Designee for the State of Florida.

A001

3/3

Rec'd w/check $150.00 #6332

8502200344 850208
PDR ADOCK 05000250
P PDR

EXHIBIT NO. 21

ccordance with 10 CFR 170.12(c), a check for $150 is attached.

attached is an evaluation of the proposed action in light of the three standards contained in 10 CFR 50.92 (No Significant Hazards).

ery truly yours,

. W. Williams, Jr.
roup Vice President
uclear Energy

W/SAV/js

ttachment

cc: J. Nelson Grace, Region II
Harold F. Reis, Esquire
Lyle Jerrett, Ph.D., Director
Office of Radiation Control
Dept. of Health & Rehabilitative Services
1323 Winewood Boulevard
Tallahassee, FL 32301

PNS-LI-85-058-1

REACTOR MATERIAL SURVEILLANCE PROGRAM

Each Type I capsule contains 28 Charpy V-notch specimens, ten Charpy specimens machined from each of the two shell forgings. The remaining eight Charpy specimens are machined from correlated monitor material. In addition, each Type I capsule contains four tensile specimens (two specimens from each of the two shell forgings) and six WOL specimens (three specimens from each of the two shell forgings). Dosimeters of copper, nickel, aluminum-cobalt, and cadmium-shielded aluminum-cobalt wire are secured in holes drilled in spacers at the top, middle, and bottom of each Type I capsule.

Each Type II capsule contains 32 Charpy V-notch specimens: eight specimens machined from one of the shell forgings, eight specimens of weld metal and eight specimens of HAZ metal, the remaining eight specimens are correlation monitors. In addition, each Type II capsule contains four tensile specimens and four WOL specimens: two tensile specimens and two WOL specimens from one of the shell forgings and the weld metal. Each Type II capsule contains a dosimeter block at the center of the capsule. Two cadmium-oxide-shielded capsules, containing the two isotopes uranium-238 and neptunium-237, are contained in the dosimeter block. The double containment afforded by the dosimeter assembly prevents loss and contamination by the neptunium-237 and uranium-238 and their activation products. Each dosimeter block contains approximately 20 milligrams of neptunium-237 and 13 milligrams of uranium-238 contained in a 3/8-inch-OD sealed brass tube. Each tube is placed in a 1/2-inch-diameter hole in the dosimeter block (one neptunium-237 and one uranium-238 tube per block), and the space around the tube is filled with cadmium oxide. After placement of this material, each hole is blocked with two 1/16-inch aluminum spacer discs and an outer 1/8-inch-steel cover disc, which is welded in place. Dosimeters of copper, nickel, aluminum-cobalt, and cadmium-shielded aluminum-cobalt are also secured in holes drilled in spacers located at the top, middle, and bottom of each Type II capsule.

Capsule Type	Capsule Identification
I	S
II	V
II	T
I	U
II	X
I	W
I	Y
I	Z

This program combines the Reactor Materials Surveillance Program into a single integrated program which conforms to the requirements of 10CFR50 Appendices G and H.

TURKEY POINT UNITS 3 AND 4

REACTOR SURVEILLANCE MATERIAL PROGRAM

PROPOSED CHANGE TO PLANT TECHNICAL SPECIFICATIONS

Appendix H requires reactors constructed of ferritic materials have their beltline regions monitored by a surveillance program complying with ASTM E185. Appendix G defines beltline materials as shell material including welds and heat affected zones, plates or forgings, that directly surround the effective height of the fuel element assemblies.

The existing Turkey Point 3 and 4 surveillance programs contain two types of surveillance capsules: 5 Type I capsules contain forging samples only; 3 Type II capsules contain forging, weld, and haz samples.

The first Type II capsule removed has defined the most limiting material in the reactor as the girth welds based on fracture toughness requirements.

Attachment 1 is an excerpt from the PTP surveillance program. Attachment 2 shows the number and identification system of Type I and II capsules in each of the Turkey Point Vessels. As can be seen, there are only two Type II capsules remaining in each vessel. Attachment 3 shows the capsule locations.

To obtain the most meaningful results from the existing program and to update the program to current Appendix H requirements, FPL proposes to remove only Type II surveillance capsules for the remainder of plant life. This requires that 3 capsules be available for removal through the end of life. Since there are only 2 capsules available for each unit, we propose to integrate the surveillance programs as permitted by Appendix H, II, C.

The requirements of 10 CFR 50 Appendix H, II, C, are:

1) Degree of Commonality

a) Design

PTP 3 and 4 are identical in design, share identical Plant Technical Specifications and have had identical major modifications such as steam generator replacement and TMI backfit modifications. The reactor vessels were fabricated the same way by the same supplier utilizing the same materials.

Applicant Correspondence File

FPL

FLORIDA POWER & LIGHT COMPANY

April 11, 1977
L-77-113

Office of Nuclear Reactor Regulation
Attention: Mr. George Lear, Chief
Operating Reactors Branch #3
Division of Operating Reactors
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Dear Mr. Lear:

Re: Turkey Point Unit 4
Docket No. 50-251
Fracture Toughness Requirements

On April 7, 1977, a meeting was held with members of your staff to discuss the status of the Turkey Point Unit 4 reactor vessel with respect to the fracture toughness requirements of Section V.B of Appendix G to 10 CFR 50. At that meeting, we showed that the weld metal surveillance data for the Turkey Point Unit 3 reactor vessel represent not only the core midplane circumferential welds in Unit 3, but in Unit 4 as well. Data supporting this conclusion are attached.

The data show that the weldment samples from a Unit 3 surveillance capsule "T" and from both the Unit 3 and Unit 4 reactor vessels were made from the same combination of filler wire heat number and welding flux lot number. However, the weldment samples from a Unit 4 surveillance capsule "T", although containing the same filler wire heat number, used a different welding flux lot number. Therefore, the Unit 3 capsule "T" sample is more representative of the Unit 4 reactor vessel.

Irradiation data from the Unit 3 capsule was submitted to the NRC on October 19, 1976 (L-76-363). The data exhibited a shelf energy of 53 ft-lbs at a fluence of 5.7×10^{18} nvt. Accordingly, the mid-plane circumferential vessel weld in Unit 4 can be expected to maintain a shelf energy level in excess of 50 ft-lbs at the 1/4 T location until at least June 1980 at which time this location will have received a fluence of 5.7×10^{18} nvt.

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HELPING BUILD FLORIDA

In the October 19 letter, we also stated that additional re... ts were being prepared by our NSSS vendor to complete summari... the fatigue, accident, and fracture analyses for Units 3 a... . We expect to receive these additional reports in draft for... about one week, and should be able to forward them on to yo... office in approximately 6 to 8 weeks.

The evaluation discussed above supports the conclusion we presented at the April 7 meeting that an Appendix G inservi... inspection of the Unit 4 reactor vessel belt-line area nee... not be conducted until after June 1980.

Very truly yours,

Robert E. Uhrig
Vice President

REU/MAS/cpc

Attachment

cc: Mr. Norman C. Moseley, Region II
Robert Lowenstein, Esquire

AEOD/C401

LOW TEMPERATURE OVERPRESSURE EVENTS AT TURKEY POINT UNIT 4

Case Study Report
Reactor Operations Analysis Branch

Office for Analysis and Evaluation
of Operational Data

March 1984

Prepared by: Wayne D. Lanning

NOTE: This report documents results of study completed to date by the Office for Analysis and Evaluation of Operational Data with regard to a particular operational situation. The findings and recommendations do not necessarily represent the position or requirements of the responsible program office nor the Nuclear Regulatory Commission.

1.0 INTRODUCTION

Before 1979, 30 reported incidents occurred in pressurized water reactors (PWRs) where the pressure/temperature limits contained in the technical specifications for the reactor coolant system were exceeded. Most of these events occurred during reactor startup or shutdown when the reactor coolant system was in a water solid condition, i.e., no steam or gas space in the pressurizer. Overpressure events primarily resulted from the loss of letdown flow with continued charging flow, inadvertent safety injection, or a heatup transient caused by starting a reactor coolant pump with the secondary coolant system temperature higher than the primary temperature. These events were caused by either equipment malfunction or operator error.

Low temperature overpressurization (LTOP) was designated a generic issue because of the possibility of a vessel failing by the brittle fracture mechanism. This failure mode may be a consequence of a pressure transient after the vessel material toughness has been reduced due to irradiation effects (i.e., increase in nil-ductility transition temperature) while a critical size flaw exists in the vessel wall. NRC resolved the generic issue in 1979* by recommending that PWR licensees implement procedures to reduce the potential for overpressure events and install equipment modifications to mitigate such events.

Since that time, ten pressure transients have been reported. The two events at Turkey Point Unit 4 on November 28 and 29, 1981 exceeded the technical specification limit (415 psig below 355°F) by about 700 and 325 psi, respectively. The two events were designated Abnormal Occurrences by the NRC (Ref. 1). The other eight reported events were mitigated by the overpressure protection system. These two overpressure events and a significant number of events at other PWRs involving inoperable trains of the overpressure protection system prompted AEOD to initiate an evaluation of operational events with the focus primarily on Turkey Point.

The overpressure protection system and the overpressure events at Turkey Point Unit 4 are described in Sections 2 and 3. Section 4 contains the analyses and evaluation of the two events, including utility management's reaction to the events. Section 5 reviews the operational experience related to inoperable trains of the overpressure protection system at other PWRs. Section 6 evaluates the adequacy of existing LTOP technical specifications. Section 7 discusses the need for operating in a water solid condition. Section 8 lists the findings and conclusions, and Section 9 contains the AEOD recommendations based on this case study.

---

*NUREG-0224 entitled, "Reactor Vessel Pressure Transient Protection for Pressurized Water Reactors," was published in September 1978 documenting the completion of the generic activity. LTOP mitigating systems were installed in most plants beginning in 1979.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555
March 11, 1987

Docket Nos. 50-250
and 50-251

Mr. C. O. Woody, Group Vice President
Nuclear Energy Department
Florida Power and Light Company
Post Office Box 14000
Juno Beach, Florida 33408

Dear Mr. Woody:

Subject: Projected Values of Material Properties For Fracture Toughness Requirements For Protection Against Pressurized Thermal Shock Events - Turkey Point Plant, Units 3 and 4

Reference: TAC Numbers 59992 and 59993

By letter dated January 23, 1986, and supplemented on June 5, and July 7, 1986, you provided your response to the Pressurized Thermal Shock (PTS) Rule, 10 CFR 50.61 for the Turkey Point Plant, Units 3 and 4. The staff, with the assistance of our contractor Brookhaven National Laboratory (BNL), have reviewed your submittals and performed confirmatory calculations.

Based on our review and confirmatory calculations, we have determined that the material properties of the reactor vessels beltline materials, the projected fluence at the inner surface of the reactor vessels at the expiration date of the licenses and the calculated RT_{PTS} at the expiration date of the licenses (April 27, 2007) to be acceptable. The calculated RT_{PTS}, both the licensee's and our confirmatory calculations, is well below the screening criterion of 300°F for the limiting circumferential weld material at the expiration date of the licenses and is therefore in conformance with the PTS Rule. The details of our evaluation and the basis for our conclusions are included in the enclosed Safety Evaluation.

The PTS Rule requires that the projected assessment of the RT_{PTS} must be updated whenever changes in core loadings, surveillance measurements or other information (including changes in capacity factor) indicate a significant change in the projected values. This ensures that you will track the accumulated fluence for the limiting beltline materials throughout the life of the plant to verify that your assumptions remain valid. In this regard, we request that you submit a re-evaluation of the RT_{PTS} and comparison of

EXHIBIT
NO.

the predicted value in any future Pressure-Temperature submittals which are submitted as required by 10 CFR 50, Appendix G, for each of the Turkey Point Units.

This concludes our actions related to the above TAC numbers.

Sincerely,

Daniel G. McDonald Jr

Daniel G. McDonald, Senior Project Manager
PWR Project Directorate #2
Division of PWR Licensing-A

Enclosures:
As stated

cc w/enclosures:
See next page

Mr. C. O. Woody
Florida Power and Light Company

Turkey Point Plant

cc:
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Newman and Holtzinger, P.C.
1615 L Street, N.W.
Washington, DC 20036

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Miami, Florida 33102

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County Manager of Metropolitan
Dade County
Miami, Florida 33130

Resident Inspector
U.S. Nuclear Regulatory Commission
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Intergovernmental Coordination
and Review
Office of Planning & Budget
Executive Office of the Governor
The Capitol Building
Tallahassee, Florida 32301

Administrator
Department of Environmental
Regulation
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Joette Lorion
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Mr. Chris J. Baker, Plant Manager
Turkey Point Nuclear Plant
Florida Power and Light Company
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Miami, Florida 33102

Attorney General
Department of Legal Affairs
The Capitol
Tallahassee, Florida 32304

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION
REGARDING PROJECTED VALUES OF MATERIAL PROPERTIES
FOR FRACTURE TOUGHNESS REQUIREMENTS
FOR PROTECTION AGAINST PRESSURIZED THERMAL SHOCK EVENTS
FLORIDA POWER AND LIGHT COMPANY
TURKEY POINT PLANT, UNITS 3 AND 4

I. Introduction

As required by 10 CFR 50.61, "Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock" (PTS Rule) which was published in the Federal Register on July 23, 1985, the licensee for each operating pressurized water reactor "shall submit projected values of RT_{PTS} (at the inner vessel surface) of reactor vessel beltline materials by giving values from the time of submittal to the expiration date of the operating license. The assessment must specify the bases for the projection including the assumptions regarding core loading patterns. This assessment must be submitted by January 23, 1986, and must be updated whenever changes in core loadings, surveillance measurements or other information indicate a significant change in projected values."

By letters dated January 23, 1986, and supplemented on June 5 and July 7, 1986, the Florida Power and Light Company submitted information on the material properties and the fast neutron fluence (E > 1.0 MeV) on the inside surface of the reactor pressure vessel, in compliance with the requirements of 10 CFR 50.61 for the Turkey Point Plant, Units 3 and 4. The RT_{PTS} and fluence values were projected to April 27, 2007, which is the expiration date of both licensees.

II. Evaluation of The Material Aspects

The controlling beltline material from the standpoint of PTS susceptibility was identified to be intermediate-to-lower girth weld SA-1101 (weld wire heat number 71249) for both unit 3 and unit 4.

The material properties of the controlling material and the associated margin and chemistry factor were reported to be:

	Utility Submittal	Staff Evaluation
Cu (copper content, %)	0.26	0.26
Ni (nickel content, %)	0.60	0.60
I (Initial RT_{NDT}, °F)	+10	+10

	Utility Submittal	Staff Evaluation
M (Margin, °F)	--	48
CF (Chemistry Factor, °F)	--	166.8

The controlling material has been properly identified. The justifications for the copper and nickel contents and the initial RT_{NDT} are given by reference to a submittal dated February 10, 1984, which was accepted by the staff on April 26, 1984 (S.A. Varga to J.W. Williams of FPL). The justifications meet our criteria for PTS submittals. The margin has been derived from consideration of the bases for these values, following the PTS Rule, Section 50.61 of 10 CFR Part 50. Assuming that the reported values of fluence are correct, Equation 1 of the PTS rule governs, and the chemistry factor is as shown above.

III. Evaluation of the Fluence Aspects

Early studies of the PTS issue for the Turkey Point plants indicated that (a) the controlling beltline material is the intermediate-to-lower circumferential weld SA-1101 and (b) a flux reduction factor of about 4.5 should be effected for both plants to prevent them from reaching the 10 CFR 50.61 screening criteria before April 2007 (i.e., the expiration of their operating licenses). To this end the licensee implemented a flux reduction scheme based on the use of part-length absorber rods located on the assemblies on the core flats. The purpose of this review was to evaluate the effectiveness of the flux reduction measures and to evaluate the projected estimate of the peak azimuthal fluence at the end of the current license on the lower circumferential welds.

The licensee's determination of the fast flux at the lower circumferential weld is based on the DOT 4.3 discrete ordinates transport code in (r,θ) geometry. The calculations employ a nuclear data library based on the 47-neutron group BUGLE-80 (ENDF/B-IV) library, and an S_8-P_3 angular decomposition. The neutron source is obtained from PDQ-7 generated pin-wise, cycle-specific power distributions. The presence of plutonium is accounted for by a mixed U+Pu core neutron source normalization factor. The fast (E > 1.0 MeV) flux at the lower circumferential weld is then given by:

$$\Phi \text{ weld} = \phi\text{DOT}(r=\text{PV inner surface}, \theta)\ P\ (z=\text{weld elevation})$$

i.e., the DOT 4.3 (r,θ) result is multiplied by the relative axial power at the elevation of the weld (from a NODE-P calculation) to provide an estimate of the three-dimensional fast flux at that location.

The basic elements of the Brookhaven National Laboratory (BNL), our contractor's, approach for determining the fast flux at the peak wall location on the lower circumferential pressure vessel welds are summarized below:

1. Forward and/or adjoint fixed source calculations are performed with DOT-4.3 in (r,θ) and (r,z) geometries in order to determine the contributions of selected assemblies and axial zones to the $E > 1.0$ MeV flux at the lower circumferential welds, near the core major axis (the peak azimuthal location).

2. The DOT calculations employ a 16-neutron group library derived from the ENDF/B-IV based on 100-neutron group EPR library and an S_8-P_3 angular decomposition.

3. Cycle-specific source data provided by the licensee are used in conjunction with the DOT-4.3 results to synthesize the three-dimensional flux. Only assembly averaged sources are considered, and the neglected pin-wise power distributions are accounted for via a generic adjustment factor determined from an earlier study.

4. An exposure correction is applied on an assembly basis and includes the effect of plutonium on both the source normalization and the energy-dependent source spectrum.

Results for the present and projected end of license fast fluences, and resulting values of RT_{PTS} at the inner surface of the lower circumferential weld near the core major axis are given in Tables 1 and 2 for Turkey Point Units 3 and 4, respectively. The four BNL results quantify the effects of exposure (Cases 1 vs. 2 and 3 vs. 4), and the licensee vs. the BNL approaches for estimating the three-dimensional flux at the limiting location.

For Unit 3: (1) the exposure effect is worth 3.5% and 7% at present and EOL conditions, respectively; (2) the axial treatment underestimates the present fluence by ~2% and the EOL fluence by ~10%; and (3) the difference between the licensee and BNL Case 1 results is <~3%.

For Unit 4, the exposure dependent results show a similar trend relative to the cases with no exposure correction, and the different axial treatments have a smaller effect (<4%). However, comparison of the licensee results and those from Case 1 show an ~12% discrepancy (vs. <~3% for Unit 3).

It is significant that, even though the BNL results for the fluence (Case 4) are higher than those obtained by the licensee, the resultant values for RT_{PTS} are still well below the NRC screening criterion of 300°F for circumferential welds, with end of license values of 271°F and 276°F for Units 3 and 4, respectively. Therefore, we conclude that the proposed flux reduction results in a RT_{PTS} which meets the 10 CFR 50.61 criterion and is acceptable.

IV. Conclusion

Both the licensee's and our confirmatory calculations are well below the screening criteria for the limiting material at the expiration date of the licenses. The licensee has calculated a RT_{PTS} of 236°F and 233°F for Units 3 and 4, respectively. As stated in the evaluation portion of this Safety Evaluation, the staff's confirmatory calculations are higher with a RT_{PTS} of 271°F and 276°F for Units 3 and 4, respectively, for the limiting circumferential weld material to April 27, 2007, which is the expiration date of both licenses.

We therefore conclude that the Turkey Point Units 3 and 4 pressure vessels meet the toughness requirements of 10 CFR 50.61 for operation to the end of their current licenses provided that future fuel loadings continue to use the special assemblies for the reduction of the fast neutron fluence to the lower circumferential welds.

In order for the staff to confirm the licensee's projected estimated RT_{PTS} throughout the life of the Turkey Point Plant, Units 3 and 4 operating licenses, the licensee is required to submit a re-evaluation of the RT_{PTS} and comparison to the predicted value with future Pressure-Temperature submittals which are required by 10 CFR 50, Appendix G.

Date:

<u>Principal Contributors</u>:

P. N. Randall
L. Lois

TABLE 1

Present and Projected EOL Fluence (>1.0 MeV) and RT_{PTS} for Turkey Point Unit-3

Case	Present Fluence(1)	RT_{PTS}(2)	End-of-License Fluence(1)	End-of-License RT_{PTS}(2)
BNL-FP&L Axial Treatment				
1. Zero Exposure	1.31	237	2.10	262
2. Exposure Corrected	1.35	239	2.25	266
BNL 3-D Synthesis				
3. Zero Exposure	1.33	238	2.31	267
4. Exposure Corrected	1.37	240	2.47	271
FP&L	1.27	236	2.15	263

(1) Fluence (>1.0 MeV) x 10^{-19} n/cm^2

(2) RT_{PTS} from Eqn. 1 of 10CFR 50.61

TABLE 2

Present and Projected EOL Fluence (>1.0 MeV) and RT_{PTS} for Turkey Point Unit-4

Case	Present Fluence(1)	RT_{PTS}(2)	End-of-License Fluence(1)	End-of-License RT_{PTS}(2)
BNL-FP&L Axial Treatment				
1. Zero Exposure	1.33	238	2.40	269
2. Exposure Corrected	1.39	240	2.60	274
BNL-3-D Synthesis				
3. Zero Exposure	1.32	238	2.48	271
4. Exposure Corrected	1.39	240	2.70	276
FP&L	1.19	233	2.16	263

(1) Fluence (>1.0 MeV) x 10^{-19} n/cm^2

(2) RT_{PTS} from Eqn. 1 of 10CFR 50.61