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Evaluation of the Integrity of SEP Reactor Vessels

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Office of Nuclear Reactor Regulation

U.S. Nuclear Regulatory Commission

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ABSTRACT

This report includes a documented review of the integrity of the 11 reactor pressure vessels covered in the Systematic Evaluation Program. This review deals primarily with the design specifications and quality assurance programs used in the vessel construction and the status of material surveillance programs, pressure-temperature operating limits, and inservice inspection programs of the applicable plants. Several generic items such as PWR overpressurization protection and BWR nozzle and safe-end cracking also are evaluated. The 11 vessels evaluated include Dresden Units 1 and 2, Big Rock Point, Haddam Neck, Yankee Rowe, Oyster Creek, San Onofre 1, LaCrosse, Ginna, Millstone 1, and Palisades.

CONTENTS

APPENDICES

EVALUATION OF THE INTEGRITY OF SEP REACTOR VESSELS

1.0 INTRODUCTION

In March 1977, the Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission, established a Systematic Evaluation Program (SEP) Review Group. During Phase I of the SEP, this group developed a list of topics to be used as the basis for performing systematic evaluations of operating reactors (Ref. 1).

The group recommended that during Phase II of the SEP those plants for which operating licenses had been issued prior to 1969 should be reviewed. Indian Point Unit 1 and Humboldt Bay Unit 3 were omitted from this list because of the uncertainty associated with their future operation. There are six plants in this category. Subsequently, five facilities operating under Provisional Operating Licenses (POLs) were added to the program since the SEP review would efficiently accomplish conversion from POL to Full-Term Operating License. The plants under SEP review are:

Plant Name	Docket No./License No.
*Dresden Unit 1	50-10/DPR-2
*Yankee Rowe	50-29/DPR-3
*Big Rock Point	50-155/DPR-6
*San Onofre 1	50-206/DPR-13
*Haddam Neck	50-213/DPR-61
*Oyster Creek	50-219/DPR-16
Dresden Unit 2	50-237/DPR-19
Ginna	50-244/DPR-18
Millstone Unit 1	50-245/DPR-21
Palisades	50-255/DPR-20
*La Crosse	50-409/DPR-45

As a part of the SEP review, the NRC staff investigated the topic of reactor vessel integrity. The staff assessment included compliance with appropriate sections of 10 CFR Part 50 including fracture toughness, neutron irradiation, surveillance programs, pressure-temperature operating limits, inservice inspection programs, transient analysis, and vessel nozzle and safe-end flaws.

NUREG-0081, issued by the NRC in June 1976, documented a review of the integrity of reactor vessels that were not designed to ASME Code, Section III (Ref. 2). Seven reactor vessels now in SEP were evaluated in this earlier review and are indicated by an asterisk in the preceding list. This study

*Denotes vessels designed to ASME Boiler and Pressure Vessel Code, Sections I and/or VIII, and reviewed in NUREG-0081. These Codes are hereinafter referred to as the ASME Code, Section I, Section III, Section VIII, and Section XI.

was used as part of the basis for the present SEP review. The information contained in that report was expanded to include all the SEP vessels and several items not included in NUREG-0081 and was updated to January 1, 1979.

The NRC document NUREG-0081 was published in response to concerns expressed by the Advisory Committee on Reactor Safeguards on the integrity of reactor vessels not constructed of SA-533 and SA-508 steels, designed to ASME Boiler and Pressure Vessel Code, Sections I and/or VIII, and where only limited conformance to ASME Code Section XI is practical (Ref. 3). From this review it was concluded that all the pre-Section III reactor vessels, except Indian Point 1, have approximately the same degree of reliability as the vessels designed to meet Section III discussed in WASH-1318 (Ref. 4).

2.0 DESIGN

The quality level initially built into a nuclear reactor pressure vessel is a very important factor affecting the integrity of the vessel throughout its service life. This quality level is directly related to the materials and construction practices used in the manufacture of the vessel. The current requirements governing these practices are given in the ASME Code, Section III. The minimum initial quality of nuclear pressure vessels built to this section is higher than that of vessels designed solely to Sections I or VIII. The main reasons for this are summarized below.

1. For nuclear service (Section III design), the properties of materials are more strictly defined and much more attention is paid to make certain that the proper environmental conditions are considered.

2. Design practices are much more highly refined and more emphasis is placed on the careful analysis of design details in preference to reliance only on the nominal membrane stresses and gross factor of safety. Comparison between the requirements of Section III and those of Section I and Section VIII shows that the nonnuclear sections gain all of their conservatism by increased wall thickness requirements, evaluated solely for membrane loads and with a safety factor of four between design and ultimate strength values. In contrast, although Section III requires a nominal factor of only three, this section of the Code provides rules for treating design features affected by fatigue loads and other secondary loading conditions known to be major contributors to failure of vessels.

3. Section III specifies that the vessel function, design conditions, and environmental loads be carefully defined in the Owner's Design Specification so that designers can account for all anticipated conditions that the vessel may experience over its service lifetime. Nonnuclear sections do not have such a requirement, and the majority of nonnuclear vessel failures reported were associated with conditions not considered during design.

4. Fabrication and installation methods are much more carefully controlled under Section III than for most nonnuclear service applications, because of the extensive quality assurance requirements of Section III.

Only four SEP reactor vessels were designed to Section III: Dresden 2, Palisades, Ginna and Millstone 1. The other seven vessels were designed to Sections I and/or VIII. However, the

quality of these pre-Section III reactor vessels is much better than that of nonnuclear vessels designed to these sections since their design procedures were supplemented by requirements of various nuclear code cases, the Navy Code, and supplementary requirements of the vessel fabricator or vendor. The various nuclear code cases and the Navy Code used in the design of these seven nuclear vessels are discussed in Appendix N.

Comparing the seven SEP nuclear vessels designed to Sections I and/or VIII with the vessels designed to Section III with respect to the four design aspects expressed above, we conclude that the initial quality of these vessels is very close to that of Section III vessels. A summary of this comparison follows:

1. All of the pre-Section III reactor vessels were made of SA-302, Grade B, plate. SA-302 is very similar to the SA-533, Grade B, Class 1, material used for modern reactor vessels and has approximately the same tensile and toughness properties. Also, these vessels have thicker walls and therefore have lower primary stresses than a Section III designed vessel. These lower stresses reduce the probability of a brittle failure. Finally, SA-302 is an acceptable material for Code Class 1 vessels according to Section III.

2. All of these pre-Section III vessels except Dresden 1 and Yankee Rowe used the Navy Code in their design. The Navy Code requires the stress analysis to include fatigue and thermal transient effects similar to that required by Section III. One vessel, Haddam Neck, was reevaluated to the ASME Code Section III rules. It was determined that the vessel met the Code rules and in some areas even exceeded them (Ref. 5).

3. All the SEP pre-Section III vessels were designed to meet conditions they might be expected to encounter during their lifetimes, such as plant heatup and cooldown, step load transients, loss of load, and turbine trip with SCRAM.

4. The supplementary requirements of the vessel manufacturers or vendors and the various code cases utilized required quality assurance programs very similar to those specified in Section III. The inspections performed are compared to Section III requirements for each of these vessels in tabular form in the discussion of the subject plants. (See appendices.)

Appendices A through K discuss the following topics: the design criteria and materials used in vessel construction; the status of material surveillance programs, pressure-temperature operating limits, inservice inspection programs; and the resolution of various generic safety items affecting vessel integrity.

Appendix L summarizes the design and material data for these vessels. Appendix M discusses the design criteria in ASME Code Sections I and VIII. Appendix N discusses the Navy Code and the code cases used in design of these plants.

The initial quality of Dresden 2, Palisades, Ginna, and Millstone 1 is considered acceptable since these vessels were designed to ASME Code Section III (1965 Edition). Class 1 vessels, including reactor vessels, designed to the 1965 Edition of Section III, are essentially the same as vessels designed to later editions of this code. The basic philosophy in Section III has not changed from 1965 to the present day. The most important changes in Section III have been to expand it to

include design criteria for components other than the reactor vessel such as supports, valves, pumps, etc. The design criteria in the 1965 Edition were primarily concerned with the reactor vessel design. However, there have been some changes in the Code that affect the reactor vessel design that should be mentioned. The most important change is the additional requirements for fracture toughness testing and the prevention of brittle fracture introduced into the Code in 1972. Vessels designed before this criteria became effective did not perform all the tests required by it. However, because fracture toughness is so important, the staff has developed procedures for analyzing the fracture toughness of these older vessels to provide the same margin of safety as for newer vessels (Ref. 8). Another important change is the requirement to leave access to permit Section XI inservice inspections (discussed in more detail in Section 6.0 of this report). Other changes that will affect the reactor vessel to a lesser extent are the inclusion of more detailed procedures for performing quality assurance examinations, more detailed administrative requirements for quality assurance programs, and the addition of requirements for performing an analysis for faulted and emergency operating conditions.

The design criteria utilized for all the reactor vessels in SEP are considered acceptable. The design criteria used provide assurance that the initial integrity of these vessels is acceptable.

3.0 MATERIALS

Reactor vessels being built today are constructed from SA-533, Grade B, Class 1, plate material and SA-508, Class 2, forging material. ASME Code Section II contains the specifications for these materials. The specifications include requirements for chemical composition, mechanical properties, and methods for steelmaking, plate preparation, and the forging process. Additional requirements for the materials in the beltline region, including weld metal, are contained in Appendix G of ASME Code, Section III. The selection of these materials is based on their adequate strength, high resistance to unstable crack extension under load ("toughness"), and their availability in the required sizes and thicknesses. In addition, the materials must allow the production of high-quality weldments and be compatible with the stainless steel cladding.

All of the 11 SEP reactor vessels except Ginna were made of SA-302 B plate material. Ginna was made from SA-508, Class 2, forging material. Both of these materials are acceptable by today's ASME Code Section III standards. Five of these early vessels (Yankee Rowe,* Palisades, Millstone 1, Oyster Creek, and Dresden 2) were made from SA-302 B material modified by the addition of 0.4% to 0.7% nickel. The modified SA-302 B metal is identical to SA-533, Grade B, Class 1, material. The nickel is added to increase hardenability; i.e., to produce higher fracture toughness properties through the thickness of plates in the quenched and tempered condition.

The most significant difference between present day vessel materials and materials used for older vessels is the control of residual elements. It has been shown conclusively that residual copper and phosphorus impurities are highly detrimental to nuclear radiation embrittlement resistance (Ref. 6). In vessels constructed in the 1960s and early 1970s, copper content in base and weld metal often exceeded 0.30% and the phosphorus content was higher than 0.02%. For vessels being constructed today, copper and phosphorus contents are required to be maintained below specified

*The nickel content in several of the Yankee Rowe plates was only about 0.2%.

levels (generally about 0.12% for copper and 0.017% for phosphorus). The effect of this control on the material chemistry can be seen from the predicted radiation damage curves in USNRC Regulatory Guide 1.99, Revision 1. These prediction curves show that the degree of radiation damage (in terms of the drop in upper shelf Charpy energy or the increase in RT_{NDT}*) is reduced by about 50% when copper and phosphorus are reduced from 0.30% and 0.02% to 0.15% and 0.017%.

The materials used in the construction of these 11 reactor vessels are considered acceptable. They are acceptable by the current ASME CODE, Section III standards.

4.0 MATERIAL SURVEILLANCE PROGRAM

Irradiation causes a decrease in upper shelf toughness values and an increase in the ductile to brittle transition temperature of reactor vessel materials. The amount of degradation is difficult to predict since it is influenced to a large degree by the chemical composition of the material and its environment. Therefore, a material surveillance program is required to monitor the effects of irradiation on the mechanical properties of the reactor vessel materials. NRC regulations, Appendix H to 10 CFR Part 50, require that surveillance programs contain tensile and Charpy specimens from vessel beltline plate, heat-affected zone and weld material. These specimens are to be removed periodically from the vessel and tested in accordance with ASTM Specification E 185-73, "Standard Recommended Practice for Surveillance Tests for Nuclear Reactor Vessels." The base and weld metal to be included in the program should represent the material that limits the operation of the reactor during its lifetime (limiting material). The bases for selecting the limiting material are initial transition temperature, and the predicted amount of radiation damage considering chemical composition (particularly copper and phosphorus) and neutron fluence. The material surveillance programs of nuclear plants applying for a construction permit today are carefully reviewed to make certain these programs comply with Appendix H, 10 CFR Part 50. However, for older plants, such as all of the SEP plants, the surveillance programs were initiated prior to the issuance of Appendix H. Therefore, we can not expect the surveillance programs of these older vessels to completely meet all the requirements of Appendix H. The method of compliance with specific provisions of Appendix H are reviewed and determined on a case-by-case basis (Refs. 7, 8).

The most common deficiencies, or areas of nonconformance, found in the programs of older vessels are (a) Charpy specimens oriented in the strong direction, (b) insufficient number of Charpy specimens, (c) lead factors higher than three, (d) no weld material specimens, and (e) surveillance materials not including the limiting vessel material. With proper care in planning tests and analyzing the results, most of these nonconformances will not affect the validity and usefulness of the surveillance program to a great extent. For example, test results from longitudinally oriented (strong direction) specimens reduced to 65% of their value provide a conservative estimate of values expected from transversely oriented (weak direction) specimens (Ref. 8). Futhermore, weld metal, which is isotropic, is generally the limiting vessel material. Paragraph II.B of Appendix H requires that surveillance programs comply with ASTM E 185-73. This Standard Recommended Practice requires 12 Charpy V-notch specimens each from the limiting base, weld and heat affected zone (HAZ) materials. Some of the early vessels have programs with only 8 or 10 Charpy specimens. In such cases, we

*The reference nil-ductility temperature, RT_{NDT}, is the highest of the nil-ductility temperatures established from drop weight tests or 60°F less than the temperature at which the material exhibits at least 35 mils lateral expansion and 50 ft-lbs absorbed energy in Charpy V-notch tests.

recommend that tests be conducted first at relatively high temperatures to establish the upper-shelf energy* with a minimum of three data points before proceeding to lower temperature tests in the transition region. When tests are started at low temperatures with only eight specimens, upper shelf values may never be reached (Ref. 9). Appendix H requires that lead factors (the ratio of neutron flux received by specimens to the maximum flux received by the vessel inner surface) be no greater than three. The reason for this provision was the concern over the rate effect on damage. However, recent research work has indicated that varying the rate over several orders of magnitude will have little effect on the amount of radiation damage incurred (Ref. 10). Therefore, lead factors up to at least 10 appear satisfactory.

Regarding the lead factor, it should be pointed out that paragraph II.C.2 of Appendix H states that accelerated capsules are acceptable provided that the surveillance program contains the minimum number of wall capsules as specified in paragraph II.C.3. The required number of wall capsules, three to five, depends on the value of the adjusted reference temperature at end of life. In the preceding discussion, a wall capsule is defined as one with a lead factor of three or less. Since many of these older vessels had more than the required number of capsules (some as many as 20), their programs usually contained the required number of capsules with a lead factor of three or less even though some of their capsules exceeded the required lead factor of three.

The most significant deficiency found in some of the older surveillance programs is that they do not contain any weld metal or that they do not include the limiting vessel material. Programs with these nonconformances must be carefully evaluated to determine their effectiveness to predict radiation damage. In some cases it is possible to supplement the data from the deficient program with data from other surveillance programs or from research programs (Ref. 10). When data on a particular material are not available, the staff uses NRC Regulatory Guide 1.99, Revision 1, to obtain conservative predictions of radiation damage.

Although all of the SEP reactor vessels were designed and built prior to the issuance of Appendix H, 10 CFR Part 50, they all have ongoing material surveillance programs that generally conform to Appendix H with the exception of Yankee Rowe. Yankee Rowe had a surveillance program that was terminated due to a structural failure of the capsule holder fixtures. However, prior to termination, five capsules had been removed from the vessel and tested. These results are considered to be sufficient to monitor the effects of radiation on the vessel materials throughout service life. Since all of the SEP surveillance programs were established prior to the issuance of Appendix H, they all have some areas where they do not completely conform to Appendix H requirements. The areas of nonconformance for each plant are discussed in detail in Appendices A through K. From our review of these programs, we conclude that they all provide a satisfactory means of monitoring radiation damage on vessel materials throughout service lifetime.

The preceding discussion has pointed out some of the major deficiencies found in the surveillance programs of older plants. It should also be pointed out that these older programs often have areas that exceed Appendix H requirements. Many of the surveillance programs of older plants contain specimen types and materials not presently required by Appendix H. Examples of additional specimens

*Upper-shelf energy is defined as the energy to fracture Charpy specimens at temperatures in which the fracture mode is 100% shear.

are WOL (wedge opening loading) and compact tension specimens. These types of specimens are used to obtain fracture toughness properties such as K_{IC}.* Older surveillance programs also often contain specimens from a correlation monitor material. The object of performing tests on a correlation material is to correlate the test results of one program with the surveillance results of other programs, including those from test reactors. Also, many of these older programs contain more capsules (hence more total specimens) than required by Appendix H.

5.0 PRESSURE-TEMPERATURE OPERATING LIMITS

Pressure-temperature limits for reactor vessel operation provide a means of assuring vessel integrity throughout its operating life. Operation in accordance with NRC regulations will ensure that, in the normal operating range, the vessel will operate in the upper-shelf region of its material toughness. The required operating limits also provide assurance that the fracture toughness of vessel materials during heatup and cooldown transients will be adequate to prevent rapid crack propagation (brittle fracture).

Pressure-temperature limits for inservice testing, heatup and cooldown, and core operation are required to be in compliance with the rules of Appendix G to 10 CFR Part 50, "Fracture Toughness Requirements." When first published in 1971, Appendix G used a transition temperature approach to establish safe operating limits. Appendix G was revised in 1973 to require a fracture mechanics approach. The fracture mechanics approach usually gives more conservative operating limits. The fracture mechanics approach relies on a fracture mechanics characterization of the material and its stress environment. Using this characterization, the stress in any portion of the vessel, in conjunction with any assumed flaw, can be compared with the stressed-flaw tolerance of the material, a material parameter such as K_{IC}. Using this parameter, the stress in the vessel can be limited such that, in the presence of an assumed flaw size so large as to ensure detection, no rapid crack propagation can occur. Above NDT, the fracture toughness of the materials used in nuclear reactor vessels increases greatly. Thus, the crack tolerance of the material at the normal operating temperatures is high. Under this system of fracture control, prevention of rapid fracture is assured by the control of stresses and flaw sizes. For nuclear vessel materials of normal shelf fracture toughness (according to Appendix G, 10 CFR 50, a Charpy upper-shelf energy of 50 ft-lb is required), very large cracks would be required to cause the onset of rapid crack propagation at operating temperature and pressure. In regions of high local stresses, such as nozzle corners, ductile tearing could commence at smaller cracks or lower pressure but, as the tear extended into a region of lower nominal stress such as the vessel wall, rapid fracture would again require very large cracks. Appendix G also states that a smaller postulated defect may be used provided that it can be justified. For example, Appendix G states that a quantitative evaluation of the fracture toughness requirements for nozzles is not feasible at this time, but preliminary data indicate that the design defect size for nozzles (considering the combined effects of internal pressure, external loading and thermal stresses) may be a fraction of that postulated for the vessel shell. Nondestructive examination methods shall be sufficiently reliable and sensitive to detect these smaller defects.

*K_{IC} is the plane strain fracture toughness of a material.

The specific methods to calculate the pressure-temperature operating limits are contained in Appendix G to ASME Code Section III. For regions remote from discontinuities (the beltline region), the stress intensity factors calculated in the development of these operating limits are based on a postulated sharp, surface flaw penetrating to a depth of 1/4 of the vessel wall thickness and having a length 1½ times the section thickness. Since the maximum size flaw that might escape detection in a preservice or inservice inspection is much smaller than this assumed flaw size, the combination of inspections and conservative pressure-temperature limits provides a high degree of assurance for vessel integrity throughout service life. For nozzles, flanges, and shell regions near discontinuities, a smaller defect size may be used. The smaller defect size must be justified and nondestructive examination methods must be sufficiently reliable and sensitive to detect these smaller defects. The procedures to calculate the stress intensity factors for these regions provide margins of safety comparable to those required for the beltline region. Appendix G provides methods to calculate stress intensities for membrane tension stress, bending stress, and stresses resulting from thermal gradients, and lists the safety factors to be applied to these stress intensities.

Irradiation degrades material toughness causing its RT_{NDT} to increase. Since the pressure-temperature limits are based on a temperature above RT_{NDT}, these limits must be revised periodically to reflect the changes in toughness. Since the postulated flaw penetrates to 1/4 the wall thickness, the increase in RT_{NDT} is based on the fluence at the 1/4 thickness location. Increases in RT_{NDT} are usually obtained from the results of the vessel's material surveillance program. If these results are for some reason not considered applicable or valid, the staff uses Regulatory Guide 1.99, Revision 1, to obtain conservative radiation damage values.

As of January 1, 1979, all of the SEP plants have pressure-temperature operating limits that are in conformance with Appendix G, 10 CFR Part 50. The NRC staff will continue to review the results of material surveillance programs of the SEP reactor vessels as well as those of later vintage vessels. These results will be used to update the operating limits of these plants. As a minimum, the staff will require that these limits meet the requirements of Appendix G, 10 CFR Part 50. In general, the same criteria will be used to evaluate the operating limits of these older reactor vessels as will be used for later vintage vessels.

In the review of the pressure-temperature operating limits, Regulatory Guide 1.99, Revision 1, was used to evaluate the amount of radiation damage, change in RT_{NDT}, on vessel materials. This guide has been approved by NRC's Regulatory Requirements Review Committee as a backfit item. The guide was used to conservatively predict radiation damage on materials that are not included in the vessel's material surveillance program. The guide was also used to check the validity of test results from the various SEP material surveillance programs by comparing them to the predictions in the guide. No major or unexpected deviations from the Regulatory Guide predictions were noted in these reviews.

6.0 INSERVICE INSPECTION PROGRAM

Following the completion of fabrication, testing, examination, and certification in accordance with the construction rules of ASME Code Section III, and as part of the requirements of ASME Code Section XI, modern reactor vessels are required to be further examined nondestructively prior to, and as a condition for, placement into nuclear power plant service. These examinations are totally unlike those generally applied to the inspection of fossil-fueled power plant boiler drums. Inservice

inspection programs for steam boiler drums (fossil vessels), which have been formalized by the National Board Inspection Code, have evolved as a result of specific requirements imposed by state jurisdictional authorities. Visual examinations and hydrostatic tests are, in general, the extent of inspection performed in such cases. The preservice examinations required by Section XI require volumetric examination of all areas of the vessel that are subject to periodic inservice inspections. These examinations cover essentially 100% of the pressure-retaining welds. These preservice examinations provide a record of any indications, such as extremely small flaws, within the limits permitted by the allowable indication standards of Section XI. Also, the mapping of small flaws provides a basis for future comparison with the results of subsequent periodic inservice examinations to determine the flaw growth during service. Evaluation and analysis of the influence on the structural integrity of any detected change in size of flaws provide, in turn, a means to assess the safety of the vessel for continued service.

Section XI identifies the principal vessel areas and the extent of those areas that are periodically examined during the entire service lifetime of the reactor vessel. The principal areas of the vessel selected for examination are (a) those that are more highly stressed (e.g., nozzle-shell junction), (b) those components for which a representative examination sampling provides an assessment of the overall structural condition of the component materials (e.g., shell weld joints), and (c) those portions of the vessel where environmental effects and irradiation could influence the properties of the vessel materials (e.g., reactor beltline region). The inspections required by Section XI have resulted in many changes in vessel design to facilitate inspection and to provide access for the conduct of examinations. The examinations required by Section XI are required to be completed during each 10-year interval of service (inspection interval). Each inspection interval is divided into three 40-month inspection periods. A specified percentage of the total required inspections on each component is required to be performed during each inspection interval.

ASME Code Section XI requires nondestructive inservice examinations utilizing volumetric, surface, and visual examination methods. As the name implies, a visual examination is a visual observation of a component for signs of leakage, corrosion, or deterioration. Surface examinations, such as dye penetrant and magnetic particle, are used to reveal flaws at or near the surface inspected. Generally, the most effective method of detecting flaws is by volumetric methods such as ultrasonic, radiography, and eddy current examination techniques. A volumetric examination examines the entire volume of a material for flaws. Rules governing the examination procedures for the three preceding volumetric examination techniques are given in Section XI. However, Section XI does not preclude the use of other examination techniques provided they can be justified (Article IWA-2240). The most effective alternative volumetric examination method (a method not specifically described in Section XI) appears to be acoustic emission. One of the important advantages of acoustic emission is that it can be used to detect flaws in regions where lack of accessibility prevents the use of other techniques. The main disadvantages of acoustic emission are (a) that it has not been tested sufficiently to prove its effectiveness, and (b) there are no codes approved by NRC to govern its use.

Many operating plants have reactor vessels that were designed prior to the initial issuance of Section XI. The design of these vessels does not provide access to permit the conduct of all examinations now required by Section XI. Furthermore, prior to September 1, 1976, there was no requirement in NRC regulations (10 CFR Part 50) governing the inservice inspection program for plants that received a construction permit prior to January 1, 1971. Thus, the inspection programs formulated by older plants were generally nonuniform. However, on September 1, 1976, a change to

paragraph 50.55a(g) became effective which requires that, throughout the service life of a nuclear facility, the reactor vessel be inspected in accordance with editions of ASME Code Section XI and addenda that become effective to the extent practical within the limitations of design, geometry and material of construction of the vessel. Thus, the inservice inspection programs for older plants will be, as far as practical, the same as for newer plants.

Under the revised requirements of paragraph 50.55a(g), inservice examinations conducted during successive 40-month periods throughout service life will comply to the extent practical with the requirements in editions of Section XI and addenda in effect no more than 6 months prior to the start of each 40-month period. For older plants the initial period will be that period starting after September 1, 1976, based on successive periods commencing at the start of facility commercial operation. The date of the first inspection period for the SEP plants under this provision follows:

Ginna	-	9/1/76
Palisades	-	10/31/76
Dresden 1	-	3/4/77
Millstone 1	-	8/28/77
Haddam Neck	-	1/1/78
San Onofre 1	-	1/1/78
Yankee Rowe	-	3/1/78
Big Rock Point	-	9/1/78
Dresden 2	-	2/9/79
LaCrosse	-	11/1/79
Oyster Creek	-	12/79

An updated inservice inspection program in accordance with 10 CFR 50.55a(g) has been submitted for all plants in SEP. The program for Big Rock Point was originally due by mid-1978. However, NRC granted, by exemption, the licensee a 7-month delay. The program for Big Rock was submitted in December 1978 and is currently under review. The NRC staff has completed its review of updated inservice inspection programs for Ginna, Palisades, Dresden 1, Millstone 1, and San Onofre 1. Safety Evaluations, including grants for relief from performing inspections on certain components in accordance with Section XI requirements where we have determined that Code requirements are impractical, for these plants have been issued. The review of the inservice inspection programs for the other SEP plants is expected to be completed by early 1980.

Since the reactor vessels in SEP were designed prior to the initial issuance of Section XI, some inspections now required by Section XI cannot be performed because of the design geometry of the plant. The most important category where relief is requested is the Category B-A* welds in the vessel core region. The only plants in SEP that can inspect these welds are San Onofre, Haddam Neck, Ginna, and Palisades. Dresden 1 management has committed to attempt an inspection of these beltline welds. In view of the importance of these welds and the fact that many older plants cannot inspect them, it is recommended that alternative inspection techniques be investigated by NRC. The technique that appears to be most promising is acoustic emission. NRC is currently funding research programs to develop acoustic emission examination methods (Ref. 11).

*Examination Category is that used in Table IWB-2600 of Section XI.

Commercial companies are also conducting research on acoustic emission. Exxon Nuclear Company has performed an acoustic emission examination of the German KRB plant nuclear steam supply system (Ref. 12). This inspection detected three significant acoustic sources and nine minor sources. The areas of significant sources were reexamined by ultrasonic techniques and flaws were found that closely resembled those predicted by acoustic emission. Electric Power Research Institute recently initiated an acoustic emission examination of the LaSalle reactor vessel during a shop hydro test. The results of this examination were compared to those obtained during the preservice ultrasonic examination. The comparison of results indicates that acoustic emission examination methods are feasible. Dunnigan/Endevco reviewed the progress that has been made in acoustic emission monitoring of nuclear plants (Ref. 13). This report also concludes that acoustic emission examination techniques are feasible for monitoring reactor vessels. Reference 13 also lists acoustic emission examinations that have been performed on commercial nuclear power plants. The examinations performed to date on nuclear plants gives credence to the acoustic emission method. It is recommended that the NRC staff formulate a set of rules to govern acoustic emission examinations so that utilities would receive credit for examinations conducted by this method.

By the end of 1979, all reactor vessels in SEP will have updated inservice inspection programs in accordance with the provisions of 10 CFR 50.55a(g). These programs will require that the examination of all vessel components be performed in accordance with updated editions of Section XI to the maximum extent possible. These inspection programs will continue to be updated throughout service life. Examinations in accordance with these inspection programs will provide assurance that the integrity of the reactor vessels in SEP will be maintained at acceptable levels throughout service life.

7.0 GENERIC SAFETY ITEMS

The status of several generic safety items that affect the integrity of the reactor vessel are discussed in this report. These items include low upper-shelf Charpy energy, PWR overpressurization protection system, BWR feedwater nozzle cracking, BWR control rod drive return line nozzle cracking, and sensitized stainless steel safe end cracking.

7.1 Low Upper-Shelf Charpy Energy

The NRC staff has determined that some operating plants may have reactor vessels with material that will have marginal toughness after several years of operation. PWR vessels are more likely to have a low upper-shelf problem than BWR vessels because their materials are exposed to much higher fluence levels. However, since some of the pre-Section III BWR vessels are small and will see relatively high fluences, the upper-shelf energy status of all SEP vessels is being reviewed as part of the SEP review.

The technical aspect of the problem is that Appendix G, 10 CFR Part 50, specifies minimum toughness requirements for operation that are based on the assumption that the upper-shelf toughness is at least 50 ft-lbs, as measured by Charpy impact tests. Recent surveillance data indicate that the toughness of materials in some plants may fall below 50 ft-lbs after comparatively short periods of operation.

Although the 50 ft-lb minimum was derived by empirical methods, it is not considered overly conservative by most experts in the field. This level was very carefully reviewed at the time Appendix G was formulated, and is supported by the best empirical correlations available. At this time, we have only limited data that justify operation with lower toughness than is indicated by the 50 ft-lb Charpy level and, except for the smallest vessels, 45 ft-lbs is about as low as we expect to be justified with current technology. However, some justification to permit operation with Charpy upper shelf as low as 33 ft-lbs is contained in Reference 14. This justification is based on an Appendix G type of analysis and on the fact that the stress levels in some vessels are lower than those permitted in ASME Code Section III.

Appendix G, 10 CFR Part 50, describes the steps that must be taken to justify continued operation in cases of low toughness (i.e., under 50 ft-lbs). It specifies that all the following must be done:

1. Augmented inservice inspection of the reactor vessel beltline.

2. Additional fracture toughness information from different types of tests, such as K_{ID} determinations using special specimens. K_{ID} is the fracture toughness value similar to K_{IC} except that it is determined from a dynamic test.

3. A fracture mechanics analysis proving that sufficient safety margins are provided.

Category A Technical Activity No. A-11 will develop criteria to evaluate low upper shelf vessels (Ref. 15). At present it appears that only two SEP vessels (Ginna and Yankee Rowe) may have an upper-shelf Charpy energy that falls below 50 ft-lbs. For Ginna, this will not occur until at least 11 EFPY (Effective Full Power Years) of operation. By this time, Activity A-11 will be completed and Ginna will be reevaluated in accordance with the resulting criteria. It is estimated that the upper-shelf energy of plate materials in the Yankee Rowe vessel will fall to about 42 ft-lbs at end of life. However, because of the low stresses in the vessel, this value of upper shelf energy may be shown to be acceptable. This item will be reviewed again following the completion of Activity A-11, when the licensee provides submittals in accordance with Appendix G.

7.2 Overpressurization Protection System

As of January 1, 1979, PWR licensees have reported 33 incidents of reactor coolant system pressure transients in excess of the Technical Specification or Appendix G pressure-temperature limits (Ref. 16). The majority of cases have occurred during reactor startup or shutdown when the reactor coolant system was in a water solid condition. Of the 30 events, 10 reached a pressure of 1000 psig or more, 4 reached a pressure of 1500 psig or more, 3 reached a pressure of 2000 psig or more, and 1 exceeded 3000 psig. However, half of these incidents occurred prior to initial criticality of the reactor. Since there was no core decay heat or fission products, these events did not pose a potential hazard to the public health and safety. All of the pressure transients were such that fracture mechanics and fatigue calculations indicate that the reactor vessels were not damaged and that continued operation of these vessels was acceptable.

The pressure transient events reported to date have affected only pressurized water reactors (PWRs). Boiling water reactors (BWRs) never operate in a water solid condition except during some hydrostatic

tests. During cold shutdown conditions for BWRs, a letdown path is maintained through the reactor water cleanup system to remove the water added to the reactor through control rod drive seals. This flow is controlled to maintain reactor water levels within a narrow range. Thus, the upper region of the reactor vessel always contains vapor (steam) or gas (air). This provides a significant capability to accept volume surges with only small pressure changes. Also, the BWR reactor is pressurized for normal operation by heatup of the coolant and follows a water saturation pressure line. Thus, high pressures are not produced unless the vessel temperature is sufficient to satisfy Appendix G pressure-temperature limits.

Because of the relatively high frequency of these pressure transients, the NRC staff has concluded that administrative procedures and overpressure protection devices should be upgraded in an appropriate time frame to reduce the likelihood of future pressure transient events. As of January 1, 1979, all SEP PWRs have upgraded their overpressurization protection systems and submitted proposed Technical Specification changes to minimize the possibility of inadvertent pressure transients. Licensing action, including the Safety Evaluation and required Technical Specification changes, has been completed for Haddam Neck, Ginna, Palisades, and Yankee Rowe. Licensing action is expected to be completed on San Onofre by the end of 1979. However, the licensee is administratively complying with their proposed Technical Specifications that contain testing and administrative procedures to protect against an overpressurization incident.

Criteria used for the review of overpressurization protection systems in operating PWRs is outlined in NRC Report NUREG-0224 (Ref. 16). These criteria differ slightly from those used to evaluate plants undergoing a construction permit or operating license review. Differences are mainly in the areas of administrative controls, and electrical and seismic design requirements of the systems. These differences are not considered to be significant in terms of the degree of protection that will be provided.

7.3 BWR Feedwater Nozzle Cracking

The feedwater nozzles of essentially all operating BWRs inspected to date have been found to have blend radius cracks, some of which propagated through the cladding into the base metal (Refs. 17, 18). In several reactors, similar cracks were found in the nozzle bore.

The deepest cracks found to date were in the nozzle bore and were of a total depth of about 1½ inches. Analyses performed by the NRC staff, which are in agreement with those done by GE and field data from operating BWRs, indicate that the initial crack growth rate is high up to crack depths of about 1/4 to 1/2 inch. Further growth is slow but would accelerate with increasing depth. Eventually, the cracks will present a repair problem if, in removing them by grinding, the ASME Code limits on nozzle reinforcement should be exceeded. The crack depth equated with the reinforcement limit will depend on the details of nozzle dimensions (see NB-3330 in Section III, ASME Code).

Feedwater nozzle cracks are of concern to the NRC staff because (a) reactor pressure vessel integrity is extremely important to safety, (b) there are uncertainties about the rate at which the cracks might grow, (c) current nozzle repair procedures require that cracks be removed, thus removing metal from a relatively high stressed region of the reactor vessel, and (d) considerable radiation exposure is received by personnel performing inspections of the nozzle region and repairing cracks in the nozzles.

The NRC staff is in general agreement with GE as to the mechanisms responsible for crack initiation and growth. Crack initiation is believed to be the result of high cycle thermal fatigue caused by rapid fluctuations in water temperature within the vessel in the sparger-nozzle region during periods of low feedwater temperature when the flow may also be unsteady and perhaps intermittent. Once initiated, the cracks are believed to be driven deeper by the larger, relatively low frequency startup/shutdown pressure and thermal cycles. The latter result from significant changes in feedwater temperature during flood-up of the reactor vessel and when feedwater heaters are put into, or taken out of, service. During normal power operation, the plant feedwater heaters maintain the feedwater temperature at about 180°F below the reactor water temperature. At low power, when the feedwater heaters are not in service, the temperature differential can be 400°F or more. We believe that the basic cause of the thermal fatigue cracking problem is this relatively large temperature differential between cold incoming feedwater and the hot reactor vessel water during low power/flow and flood-up operations.

Because of the current incomplete status of studies and design efforts to resolve the nozzle cracking issue, and because hardware changes and other long-term remedial measures will require considerable time to implement at operating facilities, certain interim revisions in operational practice are desirable.

In general, the NRC staff has concluded that BWR facility operators should monitor feedwater temperature and flow during low power operation (Ref. 18). In addition, operating procedures should be revised to minimize rapid changes in feedwater flow and/or temperature, to minimize the duration of cold feedwater injection, to avoid conditions that may lead to inadvertent or unnecessary high pressure coolant injection (HPCI) system actuation, and to avoid the introduction of cold water from the reactor cleanup system. Reactor operators should attempt to limit the temperature differential between water entering the feedwater nozzles and the reactor vessel water to no greater than the normal differential at full power. They should also avoid feedwater temperature transients to the extent practicable. It has been demonstrated that by carefully bringing feedwater heaters into service, the magnitude of feedwater temperature transients can be significantly reduced. While these steps are not expected to eliminate the nozzle cracking problem, we believe that they should help to minimize the extent of cracking until permanent changes are made.

Inspection of SEP vessel nozzles has revealed crack depths ranging from 1/4 to 1/2 inch at Millstone 1, Dresden 2, and Oyster Creek. Because of their design, the other SEP BWRs Dresden 1, LaCrosse and Big Rock Point vessel nozzles are not subject to this type of nozzle flaw. To date, all of the SEP plants having this nozzle cracking problem have taken steps to alleviate it. All affected SEP plants have completed at least one dye-penetrant examination and have removed all cracks detected in the feedwater nozzles. Oyster Creek has replaced the sparger/sleeve with an advanced design component and has removed the nozzle cladding. Millstone 1 and Dresden 2 have installed interference-fit sleeves as an interim solution and are currently planning their future long-term resolution of this problem. This interim action by the licensees is considered satisfactory. Final resolution of this item will be made upon completion of NRC Technical Activity A-10 (Ref. 14).

7.4 Control Rod Drive Return Line Nozzle Cracking

There is usually one control rod drive (CRD) return line nozzle in BWR reactor vessels, generally located from 68 inches to 100 inches above the top of the active fuel. The return line is typically

4 inches in diameter. As early as 1974, a General Electric task force on austenitic stainless steel piping noted the large measured thermal gradient in CRD return line (CRD RL) nozzles. Based on the unexpectedly high top to bottom thermal gradients in the nozzle, particularly at low flows, crack initiation susceptibility was cited and rerouting the return line was considered. In addition, recent experience with BWR feedwater nozzles has demonstrated the occurrence of crack initiation in nozzles from thermal cycling and further suggested the need to examine CRD return line nozzles. Flaws approximately 1 inch deep (including the cladding) have been reported in the return line nozzles. GE issued Service Information Letter (SIL) No. 200 in October 1976 recommending inspection of the nozzle and rerouting of the return line (Ref. 19). This SIL was amended in March 1977 to provide for valving out the return line as an interim fix.

Dye penetrant (PT) inspections of the CRD return line nozzles to date at BWR plants have revealed cracks in the majority of plants inspected. Cracking has been observed in both the blend radius and bore regions of the CRD RL nozzle. Even though most plants have a thermal sleeve in the CRD RL nozzle, which would be expected to reduce the amount or extent of cracking, cracks have been found at plants both with and without sleeves.

The cause of crack initiation appears to be thermal fatigue, similar to that experienced with BWR feedwater nozzles. The thermal cycling results from the low temperature (50°F to 100°F) condensate water that enters the reactor vessel through the CRD RL nozzle during normal operation. Although crack initiation mechanisms for the feedwater and CRD RL nozzles apear to be the same, there is a substantial difference in the steady-state stresses that ultimately affect crack growth rates. CRD RL nozzle crack growth appears to be enhanced by the existence of a continuous large thermal gradient from the top to the bottom of the nozzle (550°F at the top, 50°F at the bottom), that yields high thermal stresses.

Effective long-term solutions to this problem require that the thermal cycling in the CRD RL nozzle be eliminated. Accordingly, the General Electric Company has made recommendations, both interim and final, involving system modifications to accomplish this goal.

The initial recommended General Electric fix involved (a) valving off the CRD return line to the reactor vessel, (b) reducing CRD RL system flow, (c) raising the exhaust water pressure to a level sufficient to permit the return water to enter the reactor vessel via leakage past the sealing rings in the control rod drives rather than via the return line, (d) adding exhaust water filters, and (e) testing the modified system to verify that the control rod drives would operate properly.

The interim General Electric system modification proposed rerouting the CRD return line in conjunction with the repair and capping of the nozzle. For BWR/2 plants, GE recommended that the return line be rerouted to the feedwater system outside the primary containment and downstream of all motor operated isolation valves. For BWR/3, 4 and 5 plants, the return line could be directed to the reactor water cleanup system downstream of the last motor-operated isolation valve. A final solution involving only cutting and capping the nozzle without reroute is still under NRC review. Although in agreement with the GE proposed interim solution for return line rerouting, the NRC staff also recommended an augmented inspection of these nozzles and the reactor vessel wall below these nozzles (Ref. 18).

Dresden 1, Big Rock Point, and LaCrosse do not have a CRD RL nozzle. Therefore, this item is not applicable to these plants. Each of the applicable SEP facilities (Millstone 1, Dresden 2, and Oyster Creek) has taken action to inspect the CRD RL nozzle. Dresden 2 is currently operating with the return line valved out. Millstone 1 is operating with the return line rerouted to the feedwater system. Oyster Creek, after inspecting and finding no crack indications, is operating with the original design configuration (thermal sleeve welded to the nozzle safe end). The above interim action is presently considered to be satisfactory. However, some further modifications may be required after completion of NRC Technical Activity A-10 (Ref. 14).

7.5 Sensitized Stainless Steel Safe Ends

Stainless steel safe ends can become sensitized during welding operations or stress relieving heat treatments. Sensitization occurs when a nonstabilized stainless steel containing over about 0.02% carbon is held for a period of time at a temperature ranging from 800°F to 1600°F. The maximum allowable carbon content in 304 stainless steel is 0.08%. For 304L stainless steel the carbon is intentionally kept low, to a maximum of 0.035%, and its resistance to stress corrosion cracking is relatively good. It should be pointed out that since no minimum carbon values are required, a 304 stainless steel can have less carbon than a 304L material. Sensitization is generally attributed to the precipitation of a complex carbide in the grain boundaries. In supplying chromium for the carbide precipitate, the chromium content in the immediate vicinity of these carbides drops below a critical limit and, therefore, the material becomes subject to severe attack by corrosive media. Chlorides and fluorides are the most important contaminants, although oxygen, low pH, and elevated temperature generally must also be present for cracking to occur. When a sensitized stainless steel is subjected to corrosive environments, cracking usually occurs in the grain boundaries (intergranular). The rate of this corrosion cracking in sensitized stainless steels increases as the applied stress is increased. In safe ends, the stresses are generally low except for regions near welds where the residual stresses may be high. Thus, the major cracking is usually found in the areas around welds.

Cracking in sensitized stainless steel material resulting from stress corrosion generally occurs only in BWRs. In PWRs, controls on the water chemistry of the primary coolant provide a noncorrosive atmosphere (Refs. 20, 21). Not only are halides monitored and controlled to low levels, but oxygen concentration is also automatically held to practically zero during plant operation by the use of hydrogen overpressure. Even with the use of boric acid, the effective pH is high. No stress corrosion cracking of either sensitized or nonsensitized stainless steel is expected under these conditions. Extensive testing experiences under operating conditions have borne this fact out. Therefore, only the BWR vessels in SEP have been reviewed in this area.

All the BWRs in SEP except Big Rock Point have taken steps to resolve the sensitized stainless steel cracking problem. The Big Rock Point primary safe ends are sensitized but the carbon content of the 304 stainless steel used was held to low values. As pointed out above, this increases its resistance to stress corrosion cracking. To date, no cracks have been detected in these safe ends. LaCrosse and Millstone 1 have replaced all their sensitized stainless steel safe ends with nonsensitized material. Dresden 2 originally had 27 sensitized stainless steel safe ends. Five of these have been replaced with nonsensitized material. The remaining 22 safe ends are inspected at each refueling outage. All the safe ends in Oyster Creek are cladded with weld overlay on their inside diameter. The sensitized safe ends on Dresden 1 are made of 304L stainless steel. Since

304L has low carbon, it has a good resistance against stress corrosion cracking. So far no cracks have been reported in the Dresden 1 safe ends. During the forthcoming decontamination outage, a complete examination of these safe ends will be performed. If any large flaws are detected in this examination, the staff will require the licensee to take other steps to resolve this issue. The above inspection and repair procedures are in accordance with the recommendations of the NRC Pipe Crack Study Group (Ref. 22).

8.0 CONCLUSIONS AND RECOMMENDATIONS

From this review, including SEP Topics V-3 (Overpressurization Protection) and V-6 (Reactor Vessel Integrity), it is concluded that the integrity of all 11 reactor vessels in SEP is currently acceptable. Safety Evaluations on the overpressurization protection systems for Haddam Neck, Ginna, Palisades, and Yankee Rowe have been issued. The review of the San Onofre system is expected to be completed by the end of 1979. The integrity of these 11 vessels was evaluated by the same criteria that are used to evaluate the latest operating reactor vessels. Acceptable inservice inspection and material surveillance programs, conservative pressure-temperature operating limits, and the use of materials having acceptable fracture toughness properties provide assurance that the integrity of such vessels will be maintained at acceptable levels throughout the remainder of their service life. Based on the test results of the Ginna and Yankee Rowe surveillance programs, we conclude that some of the beltline materials of these two reactor vessels will fall below current fracture toughness requirements prior to end of life. The Ginna vessel materials are expected to fall below current standards at about 11 to 13 EFPY. Prior to this time, the NRC staff will have completed NRC Technical Activity A-11. Based on the criteria developed from this activity, the Ginna reactor vessel will be reevaluated before its materials drop below current standards. The NRC performed a fracture analysis on the Yankee Rowe reactor vessel in accordance with paragraph V.C.3 of Appendix G, 10 CFR Part 50. Because of the low stresses in this vessel, the analysis showed that the levels of upper-shelf toughness of the Yankee Rowe vessel materials throughout service life are acceptable. This topic will be reviewed again in accordance with the criteria to be developed from Technical Activity A-11. The extent of inservice examinations on Dresden 1 have been limited by high radiation levels in the primary system. However, Dresden 1 is currently shut down for an extended period during which time the primary system is being decontaminated. Upon completion of this decontamination, an extensive examination will be performed on vessel components. The integrity of the Dresden 1 reactor vessel will be established by the results of this post-decontamination inspection. The integrity of the remaining eight reactor vessels is projected to be acceptable throughout service life. This conclusion is based on the assumption that NRC reviews of safety-related topics, such as updated inservice inspection programs, are completed, including implementation, expeditiously and properly (Ref. 23).

Finally, it is noted that many of these reactor vessels in SEP, as well as many other older operating vessels, do not have the required access to permit the conduct of all examinations required by ASME Code Section XI. For such vessels, it is recommended that acoustic emission methods be considered as an alternative examination method. Rules governing the performance of acoustic emission examinations and acceptance standards should be formulated.

Detailed conclusions and recommendations for each reactor vessel are presented in the appendices.

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APPENDICES A THROUGH K

DISCUSSIONS OF REACTOR VESSEL INTEGRITY

In Appendices A through K, the integrity of the 11 reactor vessels in SEP is discussed. Conclusions and recommendations are given for each reactor vessel. Material is complete through January 1, 1979. On several items, such as inservice inspection programs and overpressurization protection systems, the information in this report is updated beyond this date.

APPENDIX A

DRESDEN NUCLEAR POWER STATION, UNIT 1
REACTOR VESSEL

Design

The Dresden 1 nuclear steam supply system, designed by General Electric is a BWR/1 model which is a dual-cycle design (Ref. 1). In this design, the steam/water mixture flows from the reactor vessel to a steam drum. Steam from the steam drum goes into the steam turbines and then returns to the steam drum as condensate. Water is pumped from the steam drum through the secondary steam generator back to the reactor vessel. At normal operating conditions, the temperature of the coolant entering the reactor vessel is 505°F.

The reactor vessel was designed and fabricated by New York Shipbuilding Corporation in accordance with the rules of ASME Code Section I and General Electric specifications (Refs. 1, 2). In addition to the Code requirements, the following quality-enhancing practices were used (Ref. 2):

1. The vessel stress analysis was performed by the vessel fabricator, New York Shipbuilding Corporation. The analysis was independently reviewed by the reactor vendor, General Electric, for conformance to minimum Code requirements.

2. The vessel was constructed of the following materials:

    a. ASTM A-302, Grade B, material was used for the vessel heads and the cylindrical region opposite the core. The material was Charpy V-notch tested and is essentially equivalent to the SA-533, Grade B, Class 1, material being used today.

    b. Flange and nozzle forgings were constructed of ASTM A-336, Grade F-1, material. This material is similar in strength level to SA-515, Grade 70, plate material. It is slightly lower in alloy content and strength level than the SA-508, Class 2, material used for most reactor vessel forgings today and, accordingly, must be used in greater thicknesses for equivalent service.

3. All plate and forging material forming the pressure boundary of the vessel was ultrasonically inspected.

4. Surfaces of plate and forging material forming the pressure boundary were magnetic particle inspected.

5. All weld preparations of ferritic materials were magnetic particle inspected prior to deposition of weld metal.

6. Pressure boundary welds were examined progressively at various stages of completion by either magnetic particle or radiographic means.

A tabulation of fabrication quality control inspections performed for the Dresden 1 reactor vessel as compared to those required by ASME Section I and Section III is included in Table A-1.

In the beltline region, the reactor vessel has an ID of 146-inches and a shell thickness of 5¼-inches. The inside of the vessel is clad with 304 stainless steel roll-bonded to the surface with a nominal thickness of 3/8-inch. The reactor vessel operating pressure is 1000 psig and its design pressure is 1250 psig. At the end-of-life, the maximum neutron fluence on the vessel wall ID is estimated to be 4.5×10^{19} n/cm^2. The date of commercial operation is July 4, 1960. As of January 1, 1979, Dresden 1 has been in operation for about 10.5 EFPY.

Materials

The reactor vessel is comprised of a top head, a bottom head, and a cylindrical section. The top head has two circumferential welds made by the submerged arc welding process and eight longitudinal welds made by manual shielded-metal arc method (Ref. 2). The cylindrical section of the vessel is made up from a shell flange and eight shell courses. The lower head is joined to the cylindrical shell by a circumferential submerged arc weld. The eight shell courses are also joined by submerged arc welds. All the shell courses, except the second and third from the vessel top shell flange, are composed of two plates joined by two longitudinal submerged arc welds. The number 2 course, containing 16-inch nozzles, has four longitudinal manual shielded-metal arc welds; and the number 3 shell course, containing 10-inch nozzles, has two longitudinal manual shielded-metal arc welds. The number 3 shell course lies just above the reactor fuel zone area. Thus, all welds in the vessel beltline region were made by the submerged arc process. These welds were made with Oxweld 40 weld wire and Linde grade 80 flux. A chemical analysis was performed on the vessel welds but the copper content was not reported. The phosphorous content varied from 0.012% to 0.019%. However, on tests conducted by Union Carbide on Oxweld 40/Linde 80 welds, copper content varied from 0.19% to 0.27%. The upper-shelf Charpy energy of these test welds varied from 90 to 119 ft-lbs.

The vessel plate material is SA-302, Grade B, steel obtained from Lukens Steel. The mill heat treatment consisted of the following:

1. Austenitized at 1650°F-1700°F, removed from the furnace, and immediately spray quenched to 1000°F in 10 minutes.

2. Tempered at 1200°F-1250°F and air cooled.

The vessel was post-weld heat treated at 1150°F and furnace-cooled periodically during welding. The vessel material was heat treated for a minimum of 29 hours and a maximum of 50 hours.

TABLE A-1

INSPECTION AND ACCEPTANCE STANDARDS FOR DRESDEN UNIT 1

	ASME Section III		Dresden #1		ASME Section I	
Materials	Exam.*	Extent	Exam.*	Extent	Exam.*	Extent
Plates	U. T.	100% Volume	U. T.	100% Volume	N. S.	-
Forgings	U. T.	100% Volume	U. T.	100% Volume	N. S.	-
Fabrication						
Weld Grooves	M. T. or P. T.	100% Surface	M. T.	100% Surface	Visual	N. S.
Weld Joints	M. T. or P. T	100% Surface	M. T.	100% Surface	-	-
Shell and Head	R. T.	100% Volume	R. T.	100% Volume	R. T.	100% Volume
Nozzle Welds	R. T.	100% Volume	R. T.	100% Volume	R. T.	100% Volume
	M. T. or P. T.	100% Surface	M. T.	100% Surface	N. S.	-
Partial Penetration Welds						
Progressive	M. T. or P. T.	100% Surface	M. T. or P. T.	100% Surface**	Visual	N. S.
Final Surface	M. T. or P. T.	100% Surface	M. T. or P. T.	100% Surface	Visual	N. S.

*Examination Notations
R. T. - Radiography
U. T. - Ultrasonic Examination
M. T. - Magnetic Particle Examination
P. T. - Liquid Penetrant Examination
N. S. - Not specified.

**Performance of this examination could not be substantiated from review of documentation available at NRC. However, due to the fact that extensive use was made of progressive examination by M. T. or Radiographic means on the full penetration welds of the Dresden 1 vessel, it is likely that this examination was performed on the partial penetration welds.

A chemical analysis was performed on all plate materials. The maximum copper content of the plates in the beltline region is 0.17%. The maximum phosphorus content is 0.018%. Charpy tests were conducted on all plate materials. The Charpy specimens were oriented in both the longitudinal and transverse directions. The Charpy energy at a test temperature of 10°F for vessel beltline region plates varied from 30 to 45 ft-lbs for specimens oriented in the transverse direction (the weak direction). These values are considered to be acceptable.

Material Surveillance Program

The initial material surveillance program for Dresden 1 consisted of six capsules placed inside the thermal shield near the core (accelerated capsules), six capsules located outside the thermal shield near the vessel wall (wall capsules), and five capsules located in the steam drum (thermal control capsules Ref. 4). These capsules contain test samples from base material obtained from a run-out of plate material from shell course number 6. This course is opposite the vertical center of the fuel. Tensile and Charpy specimens were machined from the inner and outer surfaces of the plate in both the longitudinal and transverse orientation directions. Only two capsules, both wall capsules, contain tensile specimens (a total of 12 in the transverse direction and 12 in the longitudinal direction). The accelerated capsules contain nine Charpy specimens from each combination of orientation and surface location (total of 36 per capsule). The wall capsules contain nine Charpy specimens from most of the above combinations, but do have as few as three Charpy specimens from some of the locations. This is not considered to be a deficiency in the program, since it contains many more specimens than required by Appendix H, 10 CFR Part 50. The capsules in the steam drum contain either 10 or 12 Charpy specimens from only one of the above combinations.

Following the installation of the above capsules, additional capsules containing weld and HAZ metal specimens were installed. The weld metal was made by the manual shielded-metal arc process. Since this welding method was not used for the beltline welds, this material is not considered to be representative of the beltline weld metal in the reactor vessel. Therefore, the data obtained from irradiated tests on these specimens is only of limited value. The copper content of this weld material is 0.17% (Ref. 5).

Because this surveillance program was planned before the initial issuance of Appendix H, 10 CFR Part 50, it does not meet all the requirements of this appendix. The main areas in which the Dresden 1 program does not meet Appendix H requirements are:

1. Only two capsules contain tensile specimens. However, the total number of tensile specimens (24) in these two capsules actually exceeds the total number required by Appendix H.

2. The weld metal in this program is not considered to be representative of the vessel beltline weld metal. Thus, the irradiated behavior of the vessel weld material will be predicted from Regulatory Guide 1.99.

3. Specimens from the vessel plate material were machined from surface material instead of from material located at the 1/4 thickness. Material from the surface has slightly better toughness properties.

On the other hand, the program contains over twice as many capsules as required by Appendix H. It also contains specimens of plate material oriented in both the longitudinal and transverse directions. The program has been reviewed and is considered acceptable for monitoring the irradiated properties of the vessel plate material throughout service life.

To date, five base metal wall capsules, six base metal accelerated capsules, one weld metal wall capsule, and two weld metal accelerated capsules have been removed from the reactor vessel and tested (Refs. 4, 6, 7). The test results show that the vessel plate material is more sensitive to radiation damage than the weld metal. The wall base metal specimens received fluences from 3.4×10^{17} to 1.2×10^{19} n/cm^2. The accelerated base metal capsules were subjected to fluences from 4.7×10^{18} to 1.1×10^{20} n/cm^2. Fluences on the weld metal specimens varied from 1.0×10^{19} to 1.5×10^{20} n/cm^2. Test results show that the upper-shelf energy of base material (transverse direction) will fall to a level of 50 ft-lbs at a fluence of about 10^{20} n/cm^2. The shift in RT_{NDT} as a function of neutron fluence is very close to that predicted by Regulatory Guide 1.99, Revision 1, for a copper content of 0.2%. The results from this program also indicate that the amount of radiation damage in accelerated and wall capsules is approximately the same. Also, it is noteworthy that the results on transverse and longitudinal specimens show that the percentage decrease in upper-shelf energy is less for specimens in the transverse orientation direction. This becomes especially true at relatively high fluence levels.

There is one capsule, a wall base metal capsule, remaining in the reactor vessel. This capsule is designated as a standby capsule and thus will not be removed unless an unexpected occurrence arises that requires its removal.

Pressure-Temperature Operating Limits

By letter dated November 25, 1975 and supplemented by letters dated August 19, 1976 and October 18, 1977, Commonwealth Edison requested an amendment to incorporate changes to the Dresden 1 pressurization heatup and cooldown limitations (Ref. 8). In their submittal of November 25, 1975, the licensee proposed revised operating pressure-temperature limits for Dresden 1. These operating limits were based on the criteria contained in Appendix G, 10 CFR Part 50 and Appendix G of ASME Code Section III. From our review of these proposed operating limits, we concluded that they did not conform to all the requirements of Appendix G, 10 CFR Part 50. Therefore, Commonwealth Edison revised these limits and submitted them to NRC by letter dated October 18, 1977. These operating limits were calculated for a neutron fluence on the reactor vessel wall of 1.4×10^{19} and at the 1/4 T location of 1.0×10^{19} n/cm^2. This fluence corresponds to 2.4×10^6 MW days of operation. These revised limits were reviewed by the NRC staff. Based on the results of the Dresden 1 material surveillence program and the radiation damage predictions from Regulatory Guide 1.99, Revision 1, the staff concluded that these operating limits were acceptable. These are very conservative limits that do not permit heatup with the core critical below a temperature of 235°F. This is a relatively high temperature for BWRs. Below this temperature, heatup will be accomplished by the use of heating boilers and recirculation pumps. The maximum heatup rate by this method is slow, approximately 10°F per hour (Ref. 9). However, in view of the safety factors involved, this requirement is not considered an undue burden on the licensee.

These revised pressure-temperature operating limits are in accordance with Appendix G, 10 CFR Part 50. They were approved for operation through 2.4×10^6 MW days of operation and were incorporated into the Technical Specifications as part of License Amendment No. 26 (Ref. 10).

Inservice Inspection Program

Inservice inspections have been performed on components of the Dresden 1 reactor vessel since 1960. Inspections in the 1960s were performed mostly by visual methods and were conducted to meet Illinois State and insurance requirements. In the late 1960s, attention was focused on sensitized stainless steel safe ends. Examinations on these safe ends were performed by visual, dye penetrant and ultrasonic examination methods (Ref. 11). Pre-Section XI inspections were conducted in 1960, 1961, 1963, 1964, 1967, 1968, and 1969 (Refs. 11, 12, 13). In 1971, 1973 and 1977, inservice inspections were performed on reactor vessel components in accordance with ASME Code Section XI rules (Ref. 11). A large number of the vessel welds were examined during these examinations. Items inspected include the coolant inlet and outlet, poison, pressure tap, instrument and unloading shutdown nozzle welds, part of the dollar head weld, vessel to flange and head to flange welds, the entire head, vessel cladding and closure studs. No inspections were performed on Category B-A and on most of the Category B-B vessel welds. The only significant indications found were 28 indications on the inside of the head during the 1967 inspection. These were removed by grinding.

On October 31, 1978, Dresden 1 was shut down for an extended period, approximately 1½ years. During this outage, the primary system is being decontaminated (Ref. 14). Because the extent of previous examinations has been limited by high radiation levels, it is hoped that this decontamination program will permit the conduct of more examinations in accordance with Section XI rules.

In letters dated October 19, 1978 and November 2, 1977, Commonwealth Edison submitted a revised inservice inspection program in accordance with 10 CFR 50.55a(g) (Refs. 15, 16). This revised program is based on ASME Code Section XI, 1974 Edition and Addenda through the Summer 1975. It covers the inspection period from March 4, 1977 to July 4, 1980. Since previous inspections were limited primarily because of high radiation levels, it is difficult to determine exactly what examinations can be performed on the vessel components following decontamination. Thus, the licensee requested relief from some examinations such as Category B-A and B-B vessel welds and Category B-D and B-F nozzle welds. The basis for the relief request is that the accessibility and inspectability of these welds will be determined after the chemical cleaning. At that time, the type of inspection or alternate means of inspection will be determined, and, if necessary, the inspection program will be revised. During the past several years, the licensee has worked closely with Southwest Research Institute to prepare for the post-decontamination inspection. The latest remote ultrasonic inspection equipment is being considered for these examinations. Also, acoustic emission techniques are being considered as alternate examination methods.

The staff has completed its review of this proposed inspection program. The required changes to the Technical Specifications, including the related Safety Evaluation, were made by Amendment No. 31 dated September 28, 1979. Relief was not granted for examination of the Category B-A and B-B vessel welds. We recommended that attempts be made to examine these welds by the latest remote ultrasonic methods and/or acoustic emission techniques. If, after attempting these examinations, some are found to be not feasible, the licensee should then submit specific relief requests on such items. The staff granted relief from some of the Category B-D and B-F nozzle welds. It is recommended that the licensee conduct as much of the 10-year examination interval examinations during the post-decontamination inspection as practical.

It is believed that with the reduced radiation levels and the removal of insulation and cement blocks that most, if not all, of the vessel welds required to be inspected can be examined. Thus, this post-decontamination inspection will provide an excellent indication of the integrity of the Dresden 1 reactor vessel.

Generic Safety Items

Generic safety items applicable to Dresden 1 are low upper shelf Charpy energy and sensitized stainless steel safe ends. Because of the design of the nuclear steam supply system, the coolant enters the reactor vessel, at normal operating conditions, at a temperature of 505°F. The temperature of the vessel at these conditions is about 546°F. Thus, thermal stresses will be very low. Under transient conditions, the temperature difference between the coolant and the vessel wall is also small. Since thermal stresses are considered to be responsible for the initial crack growth in the feedwater nozzles, this problem (feedwater nozzle cracking) is not considered applicable to Dresden 1. Also, since the Dresden design does not have a control rod drive return line to the reactor vessel, the CRD RL nozzle cracking problem is not applicable to Dresden 1.

Based on the test results from the material surveillance program, it is concluded that the Dresden 1 plate materials will have acceptable fracture toughness properties throughout service life. The Dresden 1 material surveillance program does not contain any weld metal specimens that are representative of the vessel beltline welds. However, from the weld procedures used for the beltline welds and the results of tests performed on similar types of welds, it is concluded that the vessel weld materials will also have acceptable fracture toughness properties throughout service life. Therefore, the Charpy upper-shelf energies of all the reactor vessel materials are considered acceptable.

Dresden 1 has an estimated total of 65 sensitized stainless steel safe ends (Refs. 17, 18). Thirty-four of these are on the reactor vessel, 23 are on the steam drum, and 8 are on the secondary steam generators. To date, no sensitized safe ends have been replaced. All the sensitized safe ends on the reactor vessel are made of Type 304L stainless steel. This material has less carbon than Type 304 stainless steel and is less sensitive to stress corrosion cracking. In about 18 years of operation, no serious flaws in these safe ends have been detected by either leakage or inservice examinations. However, in past inservice inspections only a few of these safe ends have been examined. The inservice inspection following the decontamination program should, therefore, include the examination of most of the vessel safe ends. It is recommended that all of the sensitized stainless steel safe ends be examined during this inservice inspection. However, should this not be feasible, a minimum of one safe end in each loop or of a similar type should be examined.

Since the Dresden 1 safe ends are made from a low carbon stainless steel and no serious flaws have been detected to date, it appears that these safe ends are acceptable. However, a final review of their integrity will be performed after the NRC staff has reviewed the results of the post decontamination inservice inspection.

Recommendations and Conclusions

The Dresden 1 reactor vessel was designed to ASME Code Section I. However, the requirements of this Code were supplemented by additional quality control measures that were essentially in accordance with those required by ASME Code, Section III. The design criteria and quality

assurance controls utilized in the fabrication of this vessel provide assurance that the initial integrity of the vessel is acceptable. This conclusion is borne out by the fact that the vessel has operated for over 10 EFPY without a major malfunction or flaw indication. The primary stresses in the vessel beltline region are very low, approximately 2/3 of those permitted by Section III. Although all the tests on materials now required by ASME Code Section III were not performed, we conclude from the tests that were conducted that the vessel materials have acceptable mechanical properties. Inservice inspections have been conducted on the reactor vessel and its components since 1960. However, because of high radiation levels in the primary system, the extent of these examinations has been limited. The plant is currently shut down for an extended period. During this outage, the primary system will be decontaminated and then a complete inservice inspection will be performed on the reactor vessel. It appears that all vessel welds that are required to be examined during a 10-year interval by ASME Code Section XI rules will be examined at this time. The reactor vessel is currently operating with pressure-temperature operating limits that are in accordance with Appendix G, 10 CFR Part 50. The staff will continue to review these limits and update them to account for radiation damage on the vessel beltline materials. The amount of radiation damage on plate material will be determined from the results of tests on material surveillance specimens. Radiation damage on weld material will be conservatively predicted from Regulatory Guide 1.99. To date, 14 capsules have been removed from the vessel and tested. Specimens were subjected to a wide range of fluences that cover those expected on the vessel materials throughout service life. The weld material used for the surveillance specimens is not considered to be truly representative of the vessel weld metal. Therefore, the staff will use Regulatory Guide 1.99 to predict radiation damage on the vessel weld material. The combination of inservice inspections, conservative pressure-temperature operating limits, low vessel stresses, and the use of materials having adequate toughness properties provides assurance that the vessel integrity will be maintained at acceptable levels throughout service life. This conclusion is based on the premise that no serious flaws will be detected in the post-decontamination inservice inspection. The staff will rereview the vessel integrity following this inspection. The generic safety items applicable to Dresden 1 (low upper-shelf energy and sensitized stainless steel safe ends) have been successfully resolved and will not adversely affect the integrity of the reactor vessel.

The following recommendations are made:

1. The post-decontamination inspection should include 100% of welds requiring Section XI examination. For welds that cannot be examined by ultrasonic or dye penetrant methods, it is recommended that acoustic emission techniques be considered.

2. Following the decontamination inspection, the vessel integrity should be rereviewed based on the examination results.

3. Since vessel beltline weld material is not contained in the material surveillance capsules and the chemistry of the vessel weld metal is not known, it is recommended that samples of representative vessel weld metal be obtained during the present outage and a chemical analysis be performed on these samples. Special attention should be given to determine the copper and phosphorus content. This data will permit a more accurate determination of radiation damage from Regulatory Guide 1.99.

4. The integrity of the Dresden 1 sensitized stainless steel safe ends should be rereviewed following the post-decontamination inspection.

References (Appendix A)

1. Final Hazards Summary Report for Dresden 1, April 1, 1970.*

2. Commonwealth Edison letter (M. Turbak) to NRC (Chief, ORB-2) dated March 25, 1975.*

3. Commonwealth Edison letter (M. Turbak) to NRC (Chief, ORB-2) dated December 25, 1977.*

4. F. A. Brandt and A. Alexander, General Electric Report APED 3988 dated July 1962.*

5. General Electric Report NEDO 21708 dated October 1977.*

6. L. E. Steele and C. Z. Serpan, ASTM Technical Publication 481, December 1970, page 175. Available in public technical libraries.

7. G. Reigner and G. Henderson, General Electric Report NEDC 12585 dated May 1975.*

8. Commonwealth Edison letters (R. Bolger) to Director, NRC, dated November 25, 1975, October 18, 1977, and August 19, 1976.*

9. Commonwealth Edison letter (M. Turbak) to Director, NRR, dated June 16, 1977.*

10. NRC letter (D. Ziemann) to Commonwealth Edison dated July 10, 1978.*

11. Commonwealth Edison internal memorandum dated August 20, 1970 attached to Reference 12.**

12. Commonwealth Edison letter (M. Turbak) to NRC (K. Hoge) dated November 30, 1978.**

13. General Electric Report GEAP-3753 dated July 3, 1961.*

14. Dow Chemical Company Report DNS-D1-016 dated June 15, 1977.*

15. Commonwealth Edison letter (R. Bolger) to NRC (E. Case) dated November 2, 1977.*

16. Commonwealth Edison letter (M. Turbak) to NRC (ORB-2) dated October 19, 1978.*

17. Commonwealth Edison letter (H. Blise) to AEC (P. Morris) dated September 21, 1960.*

18. Telephone conversation R. Silver, NRC, with A. Roberts, Commonwealth Edison, on August 25, 1978.**

*Available in NRC PDR for inspection and copying for a fee.
**Available in source file for USNRC Report NUREG-0569.

<u>APPENDIX B</u>

YANKEE ROWE ATOMIC POWER PLANT
REACTOR VESSEL

Design

The nuclear steam supply system, consisting of four loops, was designed by Westinghouse. The reactor vessel was designed and fabricated by Babcock and Wilcox in accordance with ASME Code Section VIII and Westinghouse specifications (Refs. 1, 2, 3). In addition to the minimum Code requirements, the following quality enhancing factors were employed:

1. The vessel stress analysis included the following:

   a. A discontinuity analysis was performed on the vessel closure head.

   b. Calculations were performed to account for nozzle stresses due to piping forces and moments on the reactor vessel primary coolant inlet and outlet nozzles.

   c. A stress analysis of the reactor vessel support lugs was performed.

   d. A stress analysis was performed to establish the allowable heatup and cooldown rates for the vessel.

2. The vessel was constructed of the following materials:

   a. SA-302, Grade B, modified slightly in chemistry for improved toughness. This material was Charpy V-notch impact tested and is essentially equivalent to the SA-533, Grade B, Class 1, material being used today. It was used for the vessel heads and the cylindrical region opposite the core.

   b. The primary coolant nozzles are SA-182 forgings modified to meet the mechanical and chemical properties of SA-302, Grade B. The material was Charpy V-notch impact tested and is very similar to the SA-503, Class 3, material being used in some vessels today.

   c. The vessel flanges are SA-105, Grade II, material and were Charpy V-notch impact tested. This material is similar to SA-508, Class 1, material. It is slightly lower in alloy content and lower in strength than the SA-508. Class 2 material being used for most reactor vessel flanges today and accordingly must be used in greater thicknesses than SA-508, Class 2, for equivalent service.

3. All plate forging material forming the pressure boundary of the vessel was ultrasonically inspected.

4. All weld preps of ferritic materials were magnetic particle inspected prior to deposition of weld metal.

5. The surfaces of all completed ferritic welds were magnetic particle inspected.

A tabulation of fabrication quality control inspections performed for the Yankee Rowe reactor vessel, as compared to those required by ASME Section VIII and Section III, is included in Table B-1.

In the beltline region the vessel ID is 109 inches and the thickness is 7-7/8 inches. The operating pressure is 2000 psig and the design pressure is 2500 psig. The maximum estimated end-of-life fluence on the vessel wall ID is 2.5×10^{19} n/cm^2. The date of commercial operation is July 1961. As of January 1, 1979, Yankee Rowe has operated for about 13 EFPY.

Materials

The hemispherical bottom head of the reactor vessel is formed of carbon steel plate clad on the inside with a sheet of 304 stainless steel. The cylindrical shell section of the reactor vessel, except for the upper shell, is made up of two rolled courses of carbon steel plate and is clad on the inside with sheets of 304 stainless steel. Each rolled course has a longitudinal weld seam and the two courses are joined together by a circumferential weld seam. Twenty equally spaced brackets for supporting the thermal insulation are welded to the outside of the lower shell section. The cylindrical upper shell section of the reactor vessel is made up of three formed segments of carbon steel plate clad on the inside with weld deposited 308L stainless steel. The three formed segments are joined together by three longitudinal weld seams. The stainless steel sheet cladding is 0.109 inch thick and was attached to the base material by spot welding. The weld deposited cladding is 1/4-inch thick.

The plates used to construct the vessel are made of SA-302B steel modified by the addition of about 0.2% nickel (Ref. 4). This is slightly less than the 0.3% to 0.7% amount required for SA-533 material. However, it is expected that this nickel additive will increase the hardenability of the material. The copper and phosphorus content of these plates varies from 0.18% to 0.20% and 0.012% to 0.02%, respectively. These values are not excessively high, and therefore these materials are expected to provide adequate resistance to radiation damage. Welds were made by the submerged arc process. The type of weld wire and flux used is unknown. No chemical analysis was made on weld material.

For both plate and weld material, RT_{NDT} is estimated to be 10°F by the procedures outlined in Standard Review Plan, Section 5.3.2. Charpy tests were performed on all vessel plate and weld materials at a test temperature of 10°F. The Charpy energy varied from 36 to 41 ft-lbs for the beltline weld metals and were about 32 ft-lbs for plate materials (strong direction).

TABLE B-1

INSPECTION AND ACCEPTANCE STANDARDS FOR YANKEE ROWE

	ASME Section III		Yankee Rowe		ASME Section VIII	
Materials	Exam.*	Extent	Exam.*	Extent	Exam.*	Extent
Plates	U.T.	100% Volume	U.T.	100% Volume	N.S.	-
Forgings	U.T.	100% Volume	U.T.	100% Volume	N.S.	-
Fabrication						
Weld Grooves	M.T. or P.T.	100% Surface	M.T.	100% Surface	Visual	N.S.
Weld Joints	M.T. or P.T.	100% Surface	M.T.	100% Surface	-	-
Shell and Head	R.T.	100% Volume	R.T.	100% Volume	R.T.	100% Volume
Nozzle Welds	R.T.	100% Volume	R.T.	100% Volume	R.T.	100% Volume
	M.T. or P.T.	100% Surface	M.T.	100% Surface	N.S.	-
Partial Penetration Welds						
Progressive	M.T. or P.T.	100% Surface	M.T. or P.T.	100% Surface*	Visual	N.S.
Final Surface	M.T. or P.T.	100% Surface	M.T. or P.T.	100% Surface	Visual	N.S.

*Examination Notations
R.T. - Radiography
U.T. - Ultrasonic Examination
M.T. - Magnetic Particle Examination
P.T. - Liquid Penetrant Examination
N.A. - Not Available; performance of this examination could not be substantiated from review of documentation available at NRC.
N.S. - Not Specified

Material Surveillance Program

The Yankee Rowe reactor pressure-vessel material surveillance program consists of two elongated specimen capsules, designated as "vessel-wall capsules," located between the thermal shield and pressure-vessel wall, and eight additional and physically similar capsules, designated "accelerated capsules," located inside the thermal shield and adjacent to the shroud surrounding the fuel core (Ref. 5). The surveillance capsules were installed in the vessel during the loading of the second core. Each surveillance capsule contains Charpy V-notch and tensile specimens from an A302-B steel fabrication test plate made from the runout of one plate for the upper vessel shell course. Charpy V-notch specimens from a 6-inch A302-B plate of a reference heat were also included in each capsule. Specimens were machined from 1/4 T material and were oriented in the longitudinal direction. No weld or HAZ specimens were included in the Yankee Rowe program. The program is based on the requirements of ASTM E 185 and also generally conforms to the requirements of Appendix H to 10 CFR Part 50. The main areas where this program does not conform to Appendix H requirements are:

1. Capsules contained only 7 Charpy specimens instead of the presently required 12.

2. The program did not contain any weld or HAZ material specimens.

3. Charpy specimens were oriented in the strong direction.

4. The program has been terminated.

Since data from capsules removed from the vessel and tested provide sufficient data to obtain transition temperature shifts and upper-shelf energy at fluences the vessel materials will be exposed to throughout service life, the low number of Charpy specimens does not detract from the program. Thus, the program has generated sufficient data to monitor the effect of radiation on vessel plate material throughout service life. Since the program did not contain samples from weld metal, the irradiated properties of the vessel weld metal will be conservatively estimated from Regulatory Guide 1.99, Revision 1.

The Yankee Rowe material surveillance program was terminated in 1965 due to structural failure of the capsule holding fixtures (Ref. 6). All capsules have been removed from the vessel. Five capsules have been tested and the remaining capsules are in storage. Because of the quality and quantity of data generated, the termination of the program is not unacceptable. From the data already obtained from the tested capsules along with the use of Regulatory Guide 1.99, the NRC staff feels that there is an adequate basis to predict the irradiated properties of the Yankee Rowe reactor vessel throughout its service life.

At the end of the second core cycle four accelerated capsules were removed from the vessel and tested (Ref. 5). At the end of the fourth core cycle one wall capsule was removed and tested (Ref. 7). All tests were conducted at the Naval Research Laboratory. The materials in these capsules were exposed to neutron fluences of 2.2×10^{18}, 5.0×10^{19}, 7.0×10^{19} and 9.0×10^{19} n/cm^2. Based on the results of these tests, we calculate that the plate material at end-of-life will have an RT_{NDT} of 280°F and an upper shelf energy (strong direction) of 52 ft-lbs at the 1/4 T location. NRL also conducted experiments on these samples to determine the effects of post-irradiation annealing on the mechanical properties. Annealing temperatures studied were 640°F, 750°F and 850°F.

Pressure-Temperature Operating Limits

In a letter dated March 15, 1976, Yankee Atomic Electric Company submitted a proposed amendment to the Yankee Rowe Technical Specifications regarding pressure-temperature operating limits (Ref. 8). Additional information on the methods used to calculate these proposed limits was supplied in letter dated June 11, 1976 (Ref. 9). Data from the material surveillance program was used to obtain irradiated values of RT_{NDT} for plate materials. Regulatory Guide 1.99 was used to estimate RT_{NDT} for weld materials. The vessel weld metal was determined to be the limiting material. Operating limit curves for inservice hydrostatic tests, heatup and cooldown, and critical core operation were calculated for the fluence estimated at the end-of-core cycle 12. The proposed Technical Specifications require that, at each refueling outage following core cycle 12, these operating limits be updated to account for further radiation damage (increase in RT_{NDT}). To predict this radiation damage, a curve is included in the specifications that plots changes in RT_{NDT} as a function of operating time. The staff approved these operating limits and incorporated them into the Technical Specifications as License Amendment No. 27 (Ref. 10). These limits were reviewed again as part of this present review and were determined to be in accordance with Appendix G, 10 CFR Part 50.

Inservice Inspection Program

An inservice inspection program in accordance with ASME Code Section XI was initiated in 1970. A preservice examination on the Yankee Rowe primary coolant system was conducted by Southwest Research Institute in accordance with the rules of the 1965 Edition of Section III. The data obtained from this inspection will be used as a baseline to which inservice examination results will be compared. Inservice inspections have been performed on the reactor vessel in 1970, 1972, 1974 and 1977 (Refs. 11, 12). These examinations generally conformed to the requirements of the 1971 Edition of Section XI. There were many welds that could not be examined because of lack of accessibility, such as Category B-A and B-B vessel welds, Category B-D nozzle to vessel welds, and Category B-F safe end welds. Items inspected include the closure head, closure head cladding, flange to vessel weld, head to flange weld, ligaments, and closure studs and nuts. The only reportable indications were several small flaws in the closure head cladding.* No inspection has been performed, or is contemplated on Category B-A vessel welds.

Several inspections were performed in the 1960s that were not in accordance with Section XI. For example, during the 1965 refueling outage, all fuel was unloaded and the core barrel and core support were removed to permit a general inspection of the vessel interior. This examination revealed that one material surveillance capsule holding fixture had broken loose from the vessel. The capsule itself was also broken and surveillance specimens were released. This debris settled in the lower head region of the vessel. The debris caused several small worn areas in the northwest quadrant of the lower head (Ref. 13). Although there were numerous scratches and gouges in the cladding, the carbon steel was exposed only in two small adjacent areas. The total area of carbon steel exposed was about 2 square inches. The defect penetrated into the base metal about 1/8 inch. The defect in the area of penetration was smooth, i.e., no sharp cracks or scratches were detected in this area.

*Components of the reactor vessel were examined during the 1978 refueling outage (October to December). Particular attention was paid to the known clad defect. The defect was examined and photographed, and no change in the size of the defect was found. (Reference YAEC letter, R. Groce, to NRC dated March 15, 1979.)

It thus appears that the defect grew by a wear type of mechanism. In the flaw region, the minimum original thickness of the base material was more than 0.3 inch greater than required by ASME Code Section III. Thus, there is still a safety margin on thickness requirements. Casts of the worn areas were made from a silicone rubber compound. These casts were used in subsequent inspections to determine if any flaw growth was occurring. So far, inspections have revealed no flaw growth (Ref. 14). Westinghouse evaluated these flaws and concluded that they do not present a safety or operational problem (Ref. 15). The staff currently agrees that these flaws present no safety hazard. However, we do recommend that the cladding in the area of these flaws be inspected for any signs of further deterioration during each inspection interval.

By letter dated December 19, 1977, Yankee Atomic Electric Company submitted a revised inservice inspection program for Yankee Rowe in accordance with the requirements of 10 CFR 50.55a(g) (Ref. 16). This revised program is for the second 40-month inspection period of the second 10-year inspection interval (March 1, 1978 to June 30, 1981). The proposed program is based on the 1974 Edition of ASME Code, Section XI and Addenda through Summer 1975. Relief from the Code examination requirements was requested for the following welds: Category B-A welds in the beltline region, some Category B-B vessel welds, some Category B-D nozzle to vessel welds, some Category B-F nozzle to safe end welds, Category B-H vessel supports, and Category B-I closure head cladding. The Category B-A and B-B welds are inaccessible due to the neutron shield tank outside the vessel and a nonremovable thermal shield inside the vessel. For the other welds that relief was requested, the licensee stated that examinations would be performed on a best effort basis. This proposed inspection program is currently being reviewed by the staff. From this SEP review, it is recommended that acoustic emission be used to examine the vessel welds that cannot be examined by ultrasonic methods.

Generic Safety Items

The generic safety items applicable to Yankee Rowe are low upper-shelf toughness and overpressurization protection. From our review of the vessel materials it is concluded that the upper-shelf energy of plate materials in the vessel beltline region is currently below 50 ft-lbs. Appendix G, 10 CFR Part 50, requires that vessel materials have an upper-shelf energy of at least 50 ft-lbs for material in the transverse (weak) orientation throughout service life. Since the vessel materials in the beltline region are subjected to high fluences, the upper-shelf energy values are expected to further degrade with operating time. Therefore, the vessel integrity, as far as low upper-shelf energy applies, will be analyzed for conditions at end-of-life. The maximum fluence on the vessel wall at the 1/4 T location is calculated to be 1.5×10^{19} n/cm^2 at the end-of-life. Based on the results of the Yankee Rowe material surveillance program, the upper-shelf energy of plate material at this fluence will be about 52 ft-lbs for material oriented in the longitudinal direction. Using the procedures outlined in NRC Standard Review Plan, Section 5.3.2, this corresponds to a value of about 35 ft-lbs for material oriented in the transverse direction.

Several abnormalities were found in this review that should be considered in the evaluation. First, the irradiated value of upper-shelf energy is based primarily on data from accelerated capsules removed from the vessel early in life. The results from accelerated capsules may give lower energy values than would be obtained on materials irradiated at slower rates. This is due to possible annealing effects that may increase toughness properties with increased exposure time. Second, during the early operation of Yankee Rowe, the operating temperature was generally below the normal temperature - often falling below 500°F (Ref. 7). Research programs have shown that

irradiation at temperatures from 450°F to 500°F produces more damage (a greater decrease in upper-shelf energy) than radiation at normal operating temperatures of about 550°F (Ref. 17). Thus, it is concluded that the amount of radiation damage on the Yankee Rowe surveillance specimens may have been augmented by these abnormalities.

Test results from research programs have also shown that the decrease in upper-shelf energy from irradiation is less for material oriented in the transverse direction than for material oriented in the longitudinal direction. This is especially true at relatively high fluence levels and when the upper-shelf energy falls below 50 ft-lbs (Ref. 18). This behavior may be due to the fact that there is a low-shelf value that the upper shelf energy will not go below regardless of the amount of radiation.

Because of the above abnormalities and test data, an attempt was made to obtain the irradiated upper-shelf energy of Yankee Rowe vessel plate material from the test results on the correlation material contained in the Yankee Rowe material surveillance program. This correlation material is SA-302, Grade B, steel. Its chemical composition is almost identical to that of the Yankee Rowe vessel plate material. Also, its mechanical properties, both irradiated and unirradiated, are very similar to the vessel materials. These properties are compared in Table B-2. The advantage to using the correlation material is that this material was tested not only in the Yankee Rowe vessel but also in a test reactor at the Naval Research Laboratory (NRL). In the NRL test program, material was irradiated and tested in both the longitudinal and transverse orientation directions. The NRL test results show that the upper-shelf energy (transverse orientation) of this correlation material drops from about 50 ft-lbs (unirradiated) to about 42 ft-lbs at a fluence of 1.5 x 10^{19} n/cm^2 (Ref. 19). The above data were obtained for irradiation at 550°F. Since this value of upper shelf energy is still below the 50 ft-lbs required by Appendix G, an evaluation of the minimum acceptable energy level for the Yankee Rowe vessel materials was made. The minimum acceptable energy for Yankee Rowe may be less than 50 ft-lbs because of the low stresses in the vessel beltline region.

This evaluation basically consisted of calculating the fracture toughness, K_{IC}, from the Charpy upper-shelf energy and performing an Appendix G, ASME Code Section III, analysis to obtain the minimum required fracture toughness (Ref. 20). The Rolfe-Novak equation was used to calculate the fracture toughness from the Charpy energy (Ref. 21). This equation is an empirical expression based on test results on many heats and types of steels. The Rolfe-Novak equation can be written:

$$K_{IC} = \left[5 S_y \left(C_v - \frac{S_y}{20}\right)\right]^{\frac{1}{2}}$$

where: S_y is the yield strength in ksi

C_v is the Charpy impact energy in ft-lbs

For a yield strength of 90 ksi, the following results are calculated:

TABLE B-2

SUMMARY OF TEST RESULTS ON THE YANKEE ROWE MATERIAL SURVEILLANCE SPECIMENS (LONGITUDINAL ORIENTATION)

Irradiation Capsule	Material	Neutron Fluence	Upper Shelf Energy (ft-lb)	Change in RT_{NDT} (°F)
Preirradiation Condition	Yankee	0.0	76	--
	ASTM	0.0	71-87	--
Vessel Wall	Yankee	2.2×10^{18}	62	110
	ASTM	2.2×10^{18}	62	85
Accelerated Capsule 8	Yankee	5×10^{19}	46	320
	ASTM	5×10^{19}	46	225
Accelerated Capsule 6	Yankee	7×10^{19}	45	360
	ASTM	7×10^{19}	44	260
Accelerated Capsule 2	Yankee	9×10^{19}	45	380
	ASTM	9×10^{19}	42	310

C_v	K_{IC}
50	143
45	135
40	126
35	117

The Appendix G analysis was made using the vessel design pressure and assuming a heatup/cooldown rate of 100°F per hour. This calculation showed that a value of 113 ksi$\sqrt{in}$ is required for K_{IC}. This value is low compared to that required for more modern vessels because of the low stresses in the Yankee Rowe vessel. From the end-of-life Charpy upper-shelf energy (42 ft-lbs) and the Rolfe-Novak correlation, it is estimated that the fracture toughness of vessel plate material at end-of-life will be approximately 130 ksi$\sqrt{in}$. Since the fracture toughness of the Yankee Rowe vessel plate material at end-of-life is well above that required by an Appendix G analysis, it is concluded that the plate material has acceptable fracture toughness properties.

Appendix G, 10 CFR Part 50, requires that all vessel materials, including welds, have an upper-shelf energy of at least 50 ft-lbs. Since the Yankee Rowe surveillance program contained no weld metal specimens, the properties of weld metal (irradiated) must be estimated from test results on unirradiated weld material. Charpy energy was obtained on vessel weld metal at one temperature, 10°F. Values varied from 36 to 41 ft-lbs. Based on test data from other surveillance programs having weld metal samples with similar unirradiated Charpy energy, it is concluded that the Yankee Rowe vessel beltline weld metal will not fall below 45 ft-lbs at a fluence of 1.5×10^{19} n/cm^2. By the same analysis used for the vessel plate material, it is concluded that this value of upper shelf energy is acceptable. Therefore, it is concluded that the Yankee Rowe reactor vessel materials will have acceptable toughness properties throughout service life.

Although no low temperature overpressurization incidents have been experienced on Yankee Rowe (Ref. 22), NRC requested that Yankee Atomic Electric Company take steps, including design modifications, necessary to preclude exceeding the limits of Appendix G, 10 CFR Part 50, during inadvertent pressure transients (Ref. 23). By letters dated December 1, 1976, May 27, 1977, and April 28, 1978, the licensee submitted to NRC plant-specific analyses in support of their proposed reactor vessel low-temperature overpressurization protection system (Ref. 24). Low-temperature overpressure protection is provided by the pressurizer solenoid operated relief valve (SORV), the pressurizer air, steam or nitrogen bubble, and the two SCS safety valves (SVS). The licensee has modified the actuation circuitry of the SORV to provide a low pressure setpoint of 500 psig. Whenever the RCS temperature is below 324°F, the SORV setpoint is manually switched to the low value (500 psig), and whenever the RSC temperature is below about 300°F, the SCS is placed in service with the accompanying two SCS SVS. A pressure transient caused by mass addition (HPSIP, LPSIP or charging pump) or heat addition (decay heat, pressurizer heaters or starting a reactor coolant pump with a hot steam generator) is terminated below the Appendix G limits by automatic operation of these valves or filling the pressurizer steam or air space. However, the mass addition from the inadvertent operation of a train of ECCS equipment (one HPSIP and one LPSIP) when the RCS temperature is between 300°F and 324°F, and the mass addition from a single LPSIP when the RCS temperature is below 200°F, results in the RCS pressure exceeding allowable limits. The licensee has proposed an equipment modification to make these events less likely (Ref. 25). The staff has completed its review of the Yankee Rowe OPS and has found it acceptable. The Technical Specifications have been revised to include testing and administrative procedures recommended by the staff. These specifications, including the related

Safety Evaluations, were incorporated into the Technical Specifications by Amendment 59 dated September 14, 1979.

Conclusions and Recommendations

The Yankee Rowe reactor vessel was designed to ASME Code Section VIII. However, the requirements of this Code were supplemented by additional quality control measures that are essentially in accordance with those required by ASME Code Section III. The design and quality control measures utilized in the fabrication of this vessel provide assurance that the initial integrity of the vessel is acceptable. This conclusion is borne out by the fact that the vessel has operated for approximately 13 EFPY without a major malfunction or flaw indication. The primary stresses in the vessel beltline region are low; about 70% of those permitted by Section III. However, in this review it was determined that the upper-shelf energy of plate materials in the vessel beltline region is below that required by Appendix G, 10 CFR Part 50. From the results of the Yankee Rowe material surveillance program and the results obtained on similar materials in research programs, it is estimated that the vessel plate material will have a Charpy upper-shelf energy of about 42 ft-lbs at end-of-life. A value of 50 ft-lbs is required by Appendix G, 10 CFR Part 50. Because of the low stresses in the vessel beltline region, it is concluded that a value of 42 ft-lbs for upper-shelf energy is acceptable. This conclusion is based on present-day technology and should be reviewed in the early 1980s as recommended below. The Yankee Rowe surveillance program was terminated in 1965 due to a structural failure. Prior to termination, five capsules were removed from the vessel and tested. Some of these capsules were subjected to fluences higher than those expected on the vessel wall ID at end-of-life. Therefore, the program is considered to have accomplished its purpose. Since the surveillance program did not contain any weld metal samples, the staff will use Regulatory Guide 1.99 to predict the irradiated properties of the vessel weld materials. Inservice inspections have been performed on components of the reactor vessel, except where limited by lack of accessibility, in accordance with ASME Code Section XI rules since 1970. We will require that future inservice examinations be performed on vessel components in accordance with Section XI rules to the maximum extent permitted by the vessel design. The reactor vessel is currently operating with pressure-temperature operating limits that are in accordance with Appendix G, 10 CFR Part 50. The staff will continue to review and update these operating limits to account for further radiation damage on the vessel beltline materials. The amount of radiation damage will be predicted from the results of tests on the Yankee Rowe surveillance specimens and the damage predictions obtained from Regulatory Guide 1.99. The combination of inservice inspections, conservative pressure-temperature operating limits, low vessel stresses, and the use of materials having adequate fracture toughness provide assurance that the vessel integrity will be maintained at acceptable levels throughout service life. The generic safety items applicable to Yankee Rowe (low upper-shelf energy and overpressurization protection) have been successfully resolved. However, since the conclusions regarding low upper-shelf energy are based on present-day technology and involve the use of empirical equations, this item should be rereviewed in the early 1980s.

The following recommendations are made:

1. Samples from several vessel welds made by the same technique as the vessel beltline welds should be taken and a chemical analysis performed on them. If possible, these samples should be taken from welds made from the same batch of weld wire and flux as the vessel beltline welds. Particular attention should be devoted to getting the copper and phosphorus content of

these samples. This data will allow the staff to more accurately predict radiation damage from Regulatory Guide 1.99.

2. Since many of the vessel welds cannot be examined in accordance with Section XI rules, it is recommended that the use of acoustic emission be considered as a means of verifying the integrity of such welds. This is considered especially important for this vessel since the upper-shelf energy of some of the vessel materials are below the requirements of Appendix G, 10 CFR Part 50.

3. The adequacy of an upper-shelf energy value of 42 ft-lbs is based on present-day technology. This item should be rereviewed upon the completion of NRC Category A Technical Activity A-11 and fracture toughness tests in the HSST program. The results of these programs should enable the staff to more accurately predict values of fracture toughness, K_{IC}, from Charpy test data. Following completion of Activity A-11, the licensee should submit a report to NRC covering the items required by Appendix G, 10 CFR Part 50, for vessels having materials under 50 ft-lbs.

References (Appendix B)

1. Yankee Nuclear Power Station FSAR, Vol. I, Section 5.**

2. Letter, Yankee Atomic Electric Company (D. Vandenberg) to NRC dated March 10, 1975.**

3. Westinghouse Specification YTR Project, Reactor Vessel, May 15, 1957.

4. Letter, Yankee Atomic Electric Company to NRC dated October 19, 1977.**

5. C. Z. Serpan, et al., NRL Report 6179, November 24, 1964.*

6. Letter, Yankee Atomic Electric Company (R. Coe) to AEC dated November 23, 1965.**

7. C. Serpan and J. Hawthorne, NRL Report 6616, September 29, 1967.*

8. Letter, YAEC (J. French) to NRC dated March 15, 1976.**

9. Letter, YAEC (J. French) to NRC dated June 11, 1976.**

10. Letter, NRC (A. Schwencer) to YAEC dated July 14, 1976.**

11 Southwest Research Institute Reports 17-2935 dated February 23, 1971, and 17-3331 dated June 1972.**

*Available at the National Technical Information Service, Springfield, VA 22161.
**Available in NRC PDR for inspection and copying for a fee.

12. Letters, YAEC (R. Groce) to NRC dated May 27, 1975 and December 1, 1977.**

13. Letter, YAEC (R. Coe) to AEC dated September 30, 1965.**

14. Letter, YAEC (L. Minnick) to AEC dated March 16, 1973.**

15. Westinghouse Report, WCAP-2855 dated October 15, 1965.**

16. Letter, YAEC (R. Groce) to NRC dated December 19, 1977.**

17. J. Hawthorne, et al., NRL Memorandum Report 1753, February 15, 1967.***

18. J. Hawthorne, ASTM Report, ASTM DS 54, 1974. Available in public technical libraries.

19. J. Hawthorne, NRL Report 7011, December 22, 1968.*

20. NRC Internal letter, L. Shao to D. Eisenhut, September 21, 1977.***

21. S. Rolfe and S. Novak, ASTM STP 514, 1970, page 124. Available in public technical libraries.

22. G. Zech, NRC Report NUREG-0224, September 1978.*

23. NRC letter (A. Schwencer) to YAEC dated August 11, 1976.**

24. Letters, YAEC (D. Vandenburgh) to NRC dated December 1, 1976, May 27, 1977, and April 28, 1978.**

25. Letter, YAEC (W. Johnson) to NRC dated April 28, 1978.**

*Available at the National Technical Information Service, Springfield, VA 22161.
**Available in NRC PDR for inspection and copying for a fee.
***Available in source file for USNRC Report NUREG-0569.

APPENDIX C

BIG ROCK POINT NUCLEAR PLANT
PRESSURE VESSEL

Design

The Big Rock Point Nuclear Plant nuclear steam supply system was designed by General Electric. It is a model BWR/1 system with a forced circulation system and an external steam drum. A steam-water mixture leaves the reactor vessel and flows to a steam drum. Saturated steam flows from the drum to the turbine. Condensate from the condenser is then pumped back to the steam drum by the feedwater pump. To complete the cycle, water in the steam drum is pumped into the reactor vessel by the recirculation pump.

The reactor vessel was designed by Combustion Engineering in accordance with General Electric specifications and the rules of ASME Code Sections I and VIII including Nuclear Code Cases 1270N, 1271N and 1273N (Ref. 1). The important supplementary requirements that exceeded the Code requirements are as follows (Ref. 2):

1. The vessel stress analysis included analysis of thermal transient and fatigue effects. The method used was based upon the method of analysis developed for Naval Reactors. The method is given in PB-15987, "Tentative Structural Design Basis for Reactor Pressure Vessels and Associated Components." The vessel stress analysis performed to the procedures outlined in this document together with the Code and Code case requirements is essentially equivalent to that required by ASME Section III for Class 1 vessels.

2. The vessel was constructed of SA-302, Grade B, plate and SA-336 forging material. These materials were Charpy V-notch impact tested. A minimum Charpy impact energy of 30 ft-lbs was required at a temperature of 10°F or lower. These materials are essentially equivalent to the SA-533, Grade B, Class 1, and SA-508, Class 2, materials being used today.

3. All forging material in the vessel pressure boundary was magnetic particle and ultrasonically inspected.

4. All stainless steel cladding was dye penetrant inspected after final stress relief. In addition, the cladding was ultrasonically inspected for bonding to the base metal.

5. The surfaces of completed pressure boundary welds were magnetic particle or liquid penetrant inspected.

6. The weld preparations in ferritic materials were magnetic particle inspected prior to deposition of weld metal.

A tabulation of material acceptance inspections for the Big Rock Point reactor vessel as compared to those required by ASME Section I and Section III is included in Table C-1.

In the beltline region, the vessel ID is 106 inches and the vessel wall thickness is 5-1/4 inches. The vessel is clad to a minimum thickness of 5/32 inch using Types 308 and 309 stainless steel welding rod. The design and operating pressures are 1715 psia and 1350 psia, respectively. The date of commercial operation is March 29, 1963. As of January 1, 1979, Big Rock Point has been in operation for almost 12 EFPY. At the end of life, the maximum value of fluence on the vessel wall ID is estimated to be 5.8×10^{19} n/cm^2.

Materials

The reactor vessel consists of a cylindrical shell with a hemispherical lower shell and a hemispherical upper shell that is removable. The cylindrical shell is made from two shell courses. Each shell course is made from two plates joined together by two longitudinal welds. The two shell courses are joined by a circumferential weld. The upper shell course is joined to a flange with another circumferential weld. The plate material is SA-302, Grade B, steel purchased from Lukens Steel (Ref. 3). All welds in the beltline region were made by the submerged metal arc process. The post-weld heat treatment was 21 hours of total stress relief treatment at 1125°F ± 25°F. The nominal chemical composition of weld metal is 0.27% copper and 0.014% phosphorus. The chemical composition of plate metal in the beltline region is 0.10% copper and 0.016% phosphorus. No drop weight tests were conducted on these materials. Charpy tests were conducted on weld and plate material at one temperature, 10°F. Plate material was tested in both the transverse and longitudinal directions. The Charpy energy for weld metal was over 50 ft-lbs, which is considered very good. The Charpy energy for plate materials varied from 27 to 30 ft-lbs in the transverse (weak) direction. These values are considered to be about average for this type of steel.

Based on chemistry and expected fluence, the limiting material is estimated to be weld metal. There is no information on the type or batch of filler metal or flux used to make the vessel welds. Therefore, at present we will consider all welds to be representive of the material surveillance weld and having the chemistry reported above. Based on data from unirradiated specimens in the material surveillance program, the initial value of RT_{NDT} of the weld material is about -50°F. The initial upper shelf energy of the weld metal is about 90 ft-lbs.

Material Surveillance Program

The material surveillance program for Big Rock Point was planned prior to the initial issuance of Appendix H, 10 CFR Part 50. The program is based on ASTM Recommended Practice E-185 dated 1964 (Ref. 4). The program consists of 12 capsules having tensile and Charpy specimens from base, HAZ, and weld materials. There are four wall capsules placed at the core midplane at positions where the core corners are closest to the vessel wall. These capsules are located close to the vessel wall where they will receive a fluence only slightly higher than the vessel wall ID. Three capsules are located inside the thermal shield at positions about 6 inches from the flat faces of the core. These accelerated capsules will see a fluence from 20 to 50 times that on the vessel wall ID. The program also includes five thermal control capsules located on top of the baffle plate. These capsules are exposed to the temperature cycles of the vessel and to a neutron flux three or four decades lower than the vessel wall. The main purpose of these specimens is to monitor any aging effect experienced by vessel materials.

TABLE C-1

INSPECTION AND ACCEPTANCE STANDARDS FOR BIG ROCK POINT

	ASME Section III		Big Rock Point		ASME Section I	
Materials	Exam.*	Extent	Exam.*	Extent	Exam.*	Extent
Plates	U.T.	100% Volume	U.T.	100% Volume	N.S.	-
Forgings	U.T.	100% Volume	U.T.	100% Volume	N.S.	-
Fabrications						
Weld Grooves	M.T. or P.T.	100% Surface	M.T.	100% Surface	Visual	N.S.
Weld Joints	M.T. or P.T.	100% Volume	M.T.	100% Surface	-	-
Shell and Head	R.T.	100% Volume	R.T.	100% Volume	R.T.	100% Volume
Welds	M.T. or P.T.	100% Surface	R.T.	100% Surface	-	-
Nozzle	R.T.	100% Volume	R.T.	100% Volume	R.T.	100% Volume
Partial Penetration Welds						
Progressive	M.T. or P.T.	100% Surface	M.T. or P.T.	100% Surface	Visual	N.S.
Final Surface	M.T. or P.T.	100% Surface	M.T. or P.T.	100% Surface	Visual	N.S.

*Examination Notations
R.T. - Radiography
U.T. - Ultrasonic Examination
M.T. - Magnetic Particle Examination
P.T. - Liquid Penetrant Examination
N.S. - Not Specified.

The capsules contain various numbers of Charpy and tensile specimens. The specimen numbers given below are for each type of material in the capsule; i.e., base, HAZ and weld material. Three wall capsules contain 12 Charpy specimens and the fourth capsule contains 9 Charpy specimens. Two accelerated capsules contain 12 Charpy specimens and the third capsule contains 8 Charpy specimens. The thermal capsules contain a total of 65 Charpy specimens of each type material. There are either two or three tensile specimens in each capsule. One accelerated and two wall capsules also contain Charpy and tensile specimens made from a standard heat of base metal from the Humboldt Bay reactor vessel. All Charpy specimens are machined from material at the 1/4 T location. Specimens from base metal are oriented in the transverse direction (the weak direction). Base metal specimens were made from left over material from two plates used to fabricate the shell courses in the beltline region of the reactor vessel. Weld metal specimens were made from a weld on these two plates that simulates a longitudinal vessel weld. The submerged metal arc process was used, but there is no record to show that the same weld wire or flux was used. The licensee is attempting to obtain this information from Combustion Engineering.

Capsules also contain nickel, iron, copper and niobium flux wire dosimeters, plus several other materials to monitor the neutron flux.

The Big Rock Point material surveillance program conforms to almost all the rules of Appendix H, 10 CFR Part 50. Some of the capsules contain less than the required number of 12 Charpy specimens for each material type. However, the program contains more than the required number of capsules and total number of specimens. Some capsules also contain only two tensile specimens instead of the required three. From our review of this program, it is concluded that it is very good and will provide sufficient data to monitor the radiation damage on the reactor vessel materials throughout their service life.

To date, five capsules have been removed from the vessel. Accelerated capsules were removed in 1964 and in 1967 (Ref. 5). Wall capsules were removed in 1964 and 1968. One thermal control capsule was removed in 1968. Tests on these surveillance specimens were conducted at the Naval Research Laboratory. The two wall capsules received fluences of 1.5 and 7.1×10^{18} n/cm^2. The two accelerated capsules received fluences of 2.3×10^{19} and 1.07×10^{20} n/cm^2. From these tests, we conclude that weld metal is the limiting vessel material. Its RT_{NDT} increases 135°F at a fluence of 7.1×10^{18} n/cm^2, and increases by 190°F at a fluence of 2.3×10^{19} n/cm^2. At the above fluence levels, the upper shelf energy of the weld metal decreases from about 90 to about 60 ft-lbs. At a fluence of 1.07×10^{20} n/cm^2, the upper shelf energy is still almost 60 ft-lbs. The shelf energy of plate material also drops to about 60 ft-lbs at a fluence of 2.3×10^{19} n/cm^2. These test results do not show any rate effect on the degree of radiation damage. Thus, the results of accelerated capsules are considered to be comparable to those of the wall capsules.

Pressure-Temperature Operating Limits

Consumers Power Company, in letters dated May 30, 1975, and June 30, 1975, submitted a request to change the Technical Specification of Big Rock Point regarding pressure-temperature operating limits (Ref. 6). A revision to the Technical Specification was required in order to comply with Appendix G, 10 CFR Part 50. The revised operating limits are based on the results of the material surveillance specimens removed from the vessel. The proposed limits were calculated for a fluence of 2.8×10^{19} n/cm^2. This fluence level will be reached by 1981 (at the 1/4 T location in the vessel

wall). The staff reviewed the proposed operating limits and concluded that they are acceptable and in conformance with Appendix G, 10 CFR Part 50 (Ref. 7). They were incorporated into the Technical Specifications as Amendment 14.

Inservice Inspection Program

Although the reactor vessel was designed and constructed prior to the initial issuance of ASME Code Section XI, an inservice inspection program based on Section XI was incorporated into the Big Rock Point Technical Specifications. This program was designated as Change No. 30 dated May 4, 1972. It was based on the 1971 Edition of Section XI plus the Summer 1971 Addenda. Exceptions to the Code requirements were primarily based on experience gained by three previous inservice inspections. The main exceptions to Code required inspections are the examination of Category B-A welds in the vessel beltline region, Category B-B welds in the vessel, Category B-D nozzle to vessel welds, and Category B-H vessel support welds.

Consumers Powers Company also stated that a device is being developed to volumetrically examine the six steam outlet nozzle to safe end welds. Previous attempts to examine these welds were unsuccessful due to the physical configuration of the nozzles. Other vessel primary nozzle to safe end welds are inaccessible because of plant design. The longitudinal and circumferential welds in the core region are not accessible because the outside of the reactor vessel is closely surrounded by concrete, and access from the inside is not feasible due to the proximity of the thermal shield. Other longitudinal and circumferential welds in the reactor vessel shell are not accessible with existing equipment. Primary nozzle to reactor vessel welds are inaccessible due to configuration of the reactor vessel internals and/or lack of equipment to perform an inspection of this nature nor does plant design permit inspection of these nozzle welds from the reactor vessel exterior.

No preservice examinations were conducted so that baseline criteria are based on the results of the first inservice examination on the particular component. A preservice type of examination was performed on the newly installed reactor depressurization system during the 1976 inservice examination. This examination was performed to the rules of the 1974 Edition of Section XI.

To date, inspections have been conducted under this program in 1973, 1974, 1975, 1976 and 1977 (Ref. 8). Examinations were performed by Southwest Research Institute and Consumers Power Company personnel. The nondestructive examinations were performed with mechanized and manual ultrasonic, liquid penetrant, and magnetic particle techniques. In this review, it was noted that many vessel welds required to the examined under Section XI rules cannot be examined in the Big Rock Point reactor vessel. However, it is also noted that there has been some improvement in this program as experience was gained from previous examinations and new inspection equipment was developed. For example, in 1972 an attempt was made to examine the vessel steam outlet nozzles using mechanized ultrasonic techniques. This examination had to be postponed because of access problems caused by the sparger ring nozzles. Modification of the inspection devise permitted a successful examination of these nozzle to vessel and nozzle safe end welds during the 1973 inspection. No indications beyond Code allowable limits have been reported. An allowable subsurface indication in a closure head nozzle weld was reported in the 1976 inspection. This area will be closely watched by NRC for any signs of flaw growth.

By letter dated July 27, 1978, Consumers Power Company submitted a proposed change to the Big Rock Point Technical Specifications to incorporate into the specifications the requirements of 10 CFR 50.55a(g) (Ref. 9). On December 22, 1978, Consumers Power Company submitted a description of their updated inspection program and specific requests for relief from Section XI requirements (Ref. 9). The proposed program is applicable for the third inspection period of the first inspection interval beginning April 1, 1979. The licensee requested relief from examining Category B-A and B-B vessel welds, some Category B-D nozzle to vessel welds, and some Category B-F safe ends. Generally, a hydrostatic leak test was proposed as an alternative examination method. It is recommended that acoustic emission be considered as an alternative examination method. The proposed program is based on the rules set forth in the 1974 Edition of Section XI including Addenda through the Summer 1975. However, it is proposed that some testing and inspection criteria meet the 1977 Edition of Section XI. Since the staff has not yet approved this edition of the Code, we requested that Consumers Power Company resubmit the program to the 1974 Edition. The revised program is expected to be submitted during the fall 1979.

Generic Safety Items

Generic safety items applicable to Big Rock Point are vessel material low upper shelf toughness and sensitized stainless steel safe ends. The feedwater nozzle and CRD RL nozzle cracking problems are not applicable to this plant. There is no CRD RL to the reactor vessel. The excess water from the control rod drive system flows into either the recirculation system or the cleanup system. The feedwater nozzles on Big Rock Point are located on the steam drum. Condensate from the turbines is pumped by the feedwater pumps to the steam drum. Water from the steam drum is pumped to the reactor vessel by the recirculation pumps. At normal operating conditions, the temperature of the water entering the vessel is 570°F. This is about 12°F lower than the vessel temperature so thermal stresses will be very low. For transient conditions the temperature differential between the inlet fluid and the vessel wall is also relatively low. Since the initial crack growth in feedwater nozzles is due to thermal stresses, Big Rock Point should have no problem regarding cracks in the recirculation nozzles on the reactor vessel (the feedwater inlet nozzles are called recirculation nozzles). To date, no flaws have been detected in the recirculation nozzles of Big Rock Point.

Big Rock Point has an excellent material surveillance program. To date, five capsules have been removed from the vessel and tested. The fluence values these capsules received are representative of those the vessel will see throughout its service life. Based on the data from the tests on the surveillance specimens, we conclude that the Big Rock Point reactor vessel materials will have adequate fracture toughness properties throughout the vessel's service life. The Charpy upper shelf energy of the vessel materials is predicted to be greater than 50 ft-lbs at end of life. Therefore, the shelf energy is in compliance with Appendix G, 10 CFR Part 50, requirements.

There are sensitized stainless steel safe ends on the Big Rock Point reactor vessel. These safe ends are made from 304 stainless steel. We requested information on these safe ends in 1970 (Ref. 10). Consumers Power Company responded with information in Reference 11. Through 1970, no flaws had been detected in these safe ends The 304 stainless steel was made with low carbon content which increases its resistance to stress corrosion cracking. Since the 1970 review of the safe ends, no flaws or cracks have been found in the sensitized safe ends. We conclude that, since the vessel has been operating for 15 years, if a corrosion problem existed there would be throughwall flaws in these safe ends by now. We also realize that inservice examinations of these safe ends

have been very limited to date. It is recommended that when the staff reviews the Big Rock Point updated inservice inspection program that is required by 10 CFR 50.55a(g), they determine if new inspection techniques and devices could increase the examination areas of those safe ends. Thus, the sensitized stainless steel safe end problem for Big Rock Point cannot be completely resolved at this time. However, from this present review it is concluded that there is no evidence of any stress corrosion cracking on these safe ends. Furthermore, we believe that there are no major flaws in these safe ends because of their low carbon content.

Conclusions and Recommendations

The Big Rock Point reactor vessel was designed to ASME Code Sections I and VIII. However, the requirements of these sections were supplemented by the requirements of Nuclear Code cases, the Navy Code and purchase specifications so that the quality control and design criteria utilized were essentially in accordance with the rules of ASME Code Section III. Therefore, the initial integrity of the vessel is considered acceptable. The primary stresses in the beltline region of the vessel are low, approximately 70% of those permitted by Section III. These low stresses, along with the use of materials with adequate fracture toughness, provide assurance that brittle fracture will not occur. Inservice examinations have been performed on components of the reactor vessel in accordance with ASME Code Section XI since 1973. However, because of lack of accessibility, many of the vessel welds could not be examined. A revised inservice inspection program was submitted to NRC in December 1978. In reviewing this program, the staff should make certain that the latest inspection technology is utilized so that examinations will be conducted to the maximum extent possible. The reactor vessel is currently operating with pressure-temperature operating limits that are in accordance with Appendix G, 10 CFR Part 50. The staff will continue to review and update these operating limits to account for further radiation damage on vessel materials. The amount of radiation damage will be determined from the results of tests on Big Rock Point's surveillance specimens. The material surveillance program has been reviewed and is considered acceptable. The combination of inservice inspections, conservative operating limits, low vessel stresses and the use of materials having adequate fracture toughness properties provides assurance that the integrity of the reactor vessel will be maintained at acceptable levels throughout service life. The generic safety items applicable to Big Rock Point (low upper shelf energy and sensitized stainless steel safe ends) have been successfully resolved and will not adversely affect the vessel integrity.

The following recommendations are made:

1. A revised inservice inspection program was submitted to NRC on December 22, 1978, and revisions to the program are expected by the fall 1979. The staff should carefully review this program to verify that examinations are proposed to the maximum extent possible. It is noted that Southwest Research Institute performed past examinations. Their personnel are very capable and have knowledge of the latest examination techniques and equipment. This knowledge should be utilized in planning future inservice inspections; for example, as it was used in planning the Dresden 1 inspection program. Acoustic emission should be considered as an examination method for areas where ultrasonic methods cannot be used.

2. It is noted that this plant has sensitized stainless steel safe ends. The staff should ensure that a sufficient number of these safe ends will be examined in future inspections to establish their integrity.

3. Acoustic emission should be considered as an alternative examination method for areas where the ultrasonic method is not feasible.

References (Appendix C)

1. Final Hazards Summary Report for Big Rock Point Plant, Sections 4 and 5, November 14, 1961.*

2. Amendment 8 to Big Rock Point, "Application for Reactor Construction Permit and Operating License."*

3. Letter, Consumers Power Company (W. Skibitsky) to Director, NRR, dated June 12, 1978.*

4. E. A. Brandt, General Electric Report GECR-4442 dated December 1963.*

5. C. C. Serpan and H. E. Watson, Nuclear Engineering and Design, Vol. II, No. 3, April 1970, p. 393. Available in public technical libraries.

6. Letters, Consumers Power Company, to Director, NRR, dated May 30, 1975 and June 30, 1975.*

7. Letter, NRC (ORB-2, D. Davis) to Consumers Power Company dated June 24, 1977.*

8. Consumers Power Company Special Reports: No. 16 dated August 20, 1973; No. 20 dated October 20, 1974; No. 22 dated August 27, 1975; No. 23 dated September 14, 1976; and No. 25 dated December 20, 1977.**

9. Letters, Consumers Power Company (D. Bixel) to Director, NRR, dated July 27, 1978 and December 22, 1978.*

10. Letter, AEC (P. Morris) to Consumers Power Company dated August 6, 1970.*

11. Letter, Consumers Power Company (R. Haueter) to AEC (P. Morris) dated September 11, 1970.*

*Available in NRC PDR for inspection and copying for a fee.
**Available in source file for USNRC Report NUREG-0569.

APPENDIX D

SAN ONOFRE NUCLEAR GENERATING STATION, UNIT 1
REACTOR VESSEL

Design

The nuclear steam supply system of San Onofre Nuclear Generating Station, Unit 1, consisting of three heat transfer loops, was designed by Westinghouse. The reactor vessel was designed and fabricated by Combustion Engineering in accordance with ASME Code Section VIII, ASME Nuclear Code Cases 1270N and 1273N, and Westinghouse specifications (Ref. 1). In addition to the minimum code requirements, the following quality enhancing factors were used (Refs. 2, 3):

1. The vessel stress analysis included analysis of thermal transient and fatigue effects. The analysis was performed in accordance with the method used in the Naval Reactors Program and given in PB-151987, "Tentative Structural Design Basis for Reactor Pressure Vessels and Directly Associated Components." The vessel stress analysis performed to the procedures outlined in this document together with the Code and Code case requirements is essentially equivalent to that required by ASME Section III for Class 1 vessels.

2. The vessel was constructed of SA-302, Grade B, plate and SA-336 forging material. These materials were Charpy V-notch and drop-weight-impact-tested and are essentially equivalent to the SA-533, Grade B, Class 1, and SA-508, Class 2, materials being used today.

3. All plate and forging material forming the pressure boundary of the vessel was ultrasonically inspected.

4. Ferritic material surfaces were magnetic particle inspected prior to being covered with weld deposit.

5. All stainless steel cladding was dye penetrant inspected after final stress relief. In areas where stainless steel clad material was weld deposited over carbon steel weld metal, dye penetrant inspection was performed both before and after final stress relief.

6. All unclad ferritic steel surfaces were magnetic particle inspected after final stress relief.

7. It was required that permanent records be kept of all fabrication inspection data.

8. In addition to the transition temperature approach, a fracture mechanics approach utilizing biaxial brittle fracture specimens has also been included to evaluate the effects of radiation on the fracture toughness of the reactor vessel materials.

The tabulation of material acceptance inspections performed for the San Onofre Unit 1 reactor vessel is compared to those required by ASME Section VIII and Section III in Table D-1.

In the beltline region the reactor vessel ID is 142 inches and its shell thickness is 9.75 inches. The inner surface of the vessel is clad with stainless steel to a minimum thickness of 0.156 inch. The design and operating pressures are 2485 psig and 2085 psig, respectively. The date of commercial operation is January 1, 1968. As of January 1, 1979, San Onofre has operated for about 8 EFPY. At the end of life the maximum fluence on the vessel wall ID is estimated to be 6.1×10^{19} n/cm^2.

Material

The reactor vessel is a cylinder with a hemispherical bottom head and a flanged and gasketed removable upper head. The cylindrical part of the vessel is composed of three shell courses: the nozzle shell, the intermediate shell and the lower shell. Each shell course is fabricated from three plates joined together by longitudinal weld seams. The three shell courses are then joined together with circumferential welds. The plate material is SA-302, Grade B, steel purchased from Lukens Steel Company. Detailed information on the vessel weld material, such as the type of weld wire and flux used, chemical composition and mechanical properties, is not available at this time. However, Southern California Edison Company is now attempting to get this information for the NRC staff.

The limiting plate materials are the three intermediate shell plates (Ref. 4). The end-of-life fluence on the ID of these plates is estimated to be 6.1×10^{19} n/cm^2. The copper content of these plates varies from 0.17% to 0.18%. The phosphorus content varies from 0.012% to 0.014%. These values are relatively low and should give these materials a good resistance to radiation damage. Based solely on fluence (no chemistry is known), the limiting weld materials are expected to be the intermediate shell longitudinal weld number 7-860A and the nozzle shell to intermediate shell circumferential weld. These two welds are expected to see fluences of 5.0 and 3.7×10^{19} n/cm^2, respectively, at the vessel wall ID at the end of life.

As stated above, the mechanical properties of the vessel weld materials are not known. The limiting plate materials have unirradiated RT_{NDT} values of 40°F, 55°F and 60°F and upper shelf Charpy energy values of 72, 76 and 79 ft-lbs. The upper shelf energy values were obtained from tests on specimens oriented in the transverse (weak) direction and machined from material at the 1/4 T location (Ref. 5). These were specially made specimens to obtain weak direction properties. Specimens in the material surveillance program were orientated in the strong direction. The upper shelf energy values for plate materials are above average values for this type of material.

TABLE D-1

INSPECTION AND ACCEPTANCE STANDARDS FOR SAN ONOFRE

	ASME Section III		San Onofre 1		ASME Section VIII	
Materials	Exam.*	Extent	Exam.*	Extent	Exam.*	Extent
Plates	U.T.	100% Volume	U.T.	100% Volume	N.S.	-
Forgings	U.T.	100% Volume	U.T.	100% Volume	N.S.	-
Fabrication						
Weld Grooves	M.T. or P.T.	100% Surface	M.T.	100% Surface	Visual	N.S.
Weld Joints	M.T. or P.T.	100% Surface	M.T. or P.T.	100% Surface	-	-
Shell and Head	R.T.	100% Volume	M.T. or P.T.	100% Volume	R.T.	100% Volume
Nozzle Welds	R.T.	100% Volume	R.T.	100% Volume	R.T.	100% Volume
	M.T. or P.T.	100% Surface	M.T. or P.T.	100% Surface	N.S.	-
Partial Penetration Welds						
Progressive	M.T or P T.	100% Surface	Visual	100% Surface	Visual	N.S.
Final Surface	M.T. or P.T.	100% Surface	M.T. or P.T.	100% Surface	Visual	N.S.

*Examination Notations
R.T. - Radiography
U.T. - Ultrasonic Examination
M.T. - Magnetic Particle Examination
P.T. - Liquid Penetrant Examination
N.S. - Not Specified

Material Surveillance Program

The San Onofre surveillance program consists of 8 capsules (5 Type I and 3 Type II capsules) located between the thermal shield and vessel wall. The capsules are attached to the thermal shield. Each Type I capsule contains 32 Charpy V-notch specimens; 8 Charpy specimens machined from each of the three San Onofre vessel intermediate shell course plates; the remaining 8 Charpy specimens are machined from correlation monitor material. In addition, each Type I capsule contains 3 tensile specimens (1 specimen from each of the three San Onofre plates) and 6 WOL specimens (2 specimens from each of the three San Onofre plates). Dosimeters of copper, Al-Co and Cd shielded Al-Co 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; 8 specimens are machined from one of the three San Onofre plates, 8 specimens of weld metal, 8 specimens of HAZ metal, and the remaining 8 specimens are made from a correlation monitor material. In addition, each Type II capsule contains 4 tensile specimens and 4 WOL specimens; 2 tensile specimens and 2 WOL specimens from one of the three San Onofre plates and the weld metal. Each Type II capsule contains a dosimeter block. Two cadmium oxide shielded capsules, one containing each of the two isotopes U^{238} and Np^{237}, are contained in the dosimeter block along with wires of Co-Al (cadmium shielded and unshielded) and nickel. Dosimeters of Al-Co and Cd shielded Al-Co will also be secured in holes drilled in spacers located at the top and bottom of each Type II capsule.

The base material was taken from the ends of the three plates used to make the intermediate shell course. The weld and heat affected zone materials were obtained from a nozzle cutout. The correlation monitor material, SA-302, Grade B, was furnished by the U.S. Steel Corporation through Subcommittee II of ASTM Committee E10 on Radioisotopes and Radiation Effects. The copper content of the weld metal is 0.19% (Ref. 4).

All specimens were machined from material at the 1/4 T location. Both Charpy and tensile specimens were oriented in the longitudinal direction (the strong direction).

Since the program was initiated prior to the issuance of Appendix H, 10 CFR Part 50, it does not comply with all the Appendix H requirements. The main areas of noncompliance are: (1) 8 Charpy specimens in lieu of the 12 required, (2) Charpy specimens of base material oriented in the strong direction, and (3) some capsules do not contain weld samples. On the positive side, it should be pointed out that the program includes more than the required number of capsules and that base material from all three intermediate shell courses is included. We conclude that this surveillance program is acceptable and that it will provide sufficient data to adequately monitor radiation damage throughout the vessel's service life.

Two capsules have been removed from the vessel and tested at Southwest Research Institute. In 1970, Capsule A (a Type II capsule) was pulled (Ref. 5). This capsule received a fluence of 1.2×10^{19} n/cm^2. Material from plate 7601-9 showed an increase in RT_{NDT} of 100°F and a drop in upper shelf energy from 100 to 80 ft-lbs. The weld material showed an increase in RT_{NDT} of only 80°F and a drop in upper shelf energy from 98 to 70 ft-lbs. In 1972, Capsule D was removed after receiving a fluence of 2.36×10^{19} n/cm^2 (Ref. 6). This is a Type I capsule and thus contains no weld metal specimens. Plate material 7601-9 had an increase in RT_{NDT} of 130°F and an upper shelf drop to 73 ft-lbs. The other two plate materials showed a very similar sensitivity to radiation

damage. Westinghouse recalculated the fluence values and determined that Capsule A had received 2×10^{19} and Capsule D had received 3.35×10^{19} n/cm^2 (Ref. 7).

A third capsule was withdrawn from the vessel in the fall of 1978. This capsule is being tested at Westinghouse. The test results are expected to be available by late 1979 or early 1980.

Pressure-Temperature Operating Limits

In a letter dated June 15, 1973, Southern California Edison Company submitted a proposed change to the Technical Specifications of San Onofre regarding pressure-temperature operating limits (Ref. 8). The limits were calculated for operation through 6 EFPY, at which time the fluence at the 1/4 T location is estimated to be 6×10^{18} n/cm^2. These limits are based on the results of tests on the material surveillance specimens. The limiting material is vessel base metal. After 6 EFPY, the Technical Specifications require that these limits be revised to account for increases in radiation damage. These proposed changes were reviewed and found acceptable. They were incorporated into the Technical Specifications as Change No. 14 (Ref. 9).

At 6 EFPY (1976) the above limits were revised as required by the Technical Specifications. The new operating limits are effective through 12 EFPY (Ref. 7). These operating limits were reviewed as part of SEP in the fall of 1978. From this review it is concluded that the San Onofre operating limits are in accordance with Appendix G, 10 CFR Part 50, and are acceptable for operation through 12 EFPY.

Inservice Inspection Program

An inservice inspection program for San Onofre was started in the late 1960s. This was prior to the initial issuance of ASME Code Section XI. This program was revised in 1973 to conform to Section XI, 1971 Edition and Addenda through Winter of 1971 (Ref. 8). The first 10-year inspection interval was scheduled to end in 1979. However, because the inspection frequency was increased to three times that required by Section XI, the first interval was completed in 1977. This augmented frequency was required while modifications were being made to the emergency core cooling system.

Inspections have been conducted on the reactor vessel in 1970, 1973, 1975 and 1976 (Ref. 10). The first two inspections were conducted by Southwest Research Institute and the last two by Babcock & Wilcox. The only reportable indications were small surface indications on four vessel closure nuts and cracks in four of the six flexure type supports for the thermal shield found in the 1976 inspection. During the 1978 refueling outage, five of these supports were reinspected and cracks were found on all but one (Ref. 11). However, it appears that there was no further crack growth on the four supports where cracks were detected in the 1976 examination. Inspections were performed on all welds requiring inspection with several minor exceptions. These are discussed later in this section in the review of the present updated inspection program. The vessel welds in the beltline region were inspected during the 1976 inspection. No reportable indications were found.

On September 28, 1977, Southern California Edison Company submitted a revised inservice inspection program to comply with 10 CFR 50.55a(g) (Ref. 12). This program is for the first inspection period of the second 10-year inspection interval. This inspection period began on January 1, 1978. The program conforms to the requirements of Section XI, 1974 Edition and Addenda through Summer 1975,

with several relatively minor exceptions. Relief was requested for the following items: (1) conduct examination of vessel support lugs at the end of the 10-year interval, (2) examine the vessel cladding patches at the end of the 10-year interval, and (3) exclude examination of the dollar plate weld in the closure head. The NRC technical staff has reviewed this program and determined it to be acceptable (Ref. 13). The Safety Evaluation was included as part of Amendment No 46. Items (1) and (2) are considered to be justifiable since they require the removal of the core barrel. Removal of the core barrel more than once per inspection interval would be an undue hardship on the licensee. Regarding item (3), the dollar plate weld is completely inaccessible due to control rod drive penetration locations and no inspection is considered feasible. An exemption is also recommended regarding the surface examination of the lower 270° of several nozzle to safe end and safe end to pipe welds because of the lack of accessibility due to the closeness of the reactor cavity shield. However, these welds will be examined 100% volumetrically.

The San Onofre inservice inspection program is considered to be very good. It is comparable to programs for vessels designed in the early 1970s.

An inservice examination was performed in 1978, but the results have not been reviewed by the staff.

Generic Safety Items

Generic safety items covered in this report that are applicable to San Onofre are low upper shelf toughness and overpressurization protection. It was also noted in this review that the primary vessel nozzles were not sensitized. The nozzle end was buttered with Inconel prior to the final post weld heat treat. A stainless steel safe end was attached after the final post-weld heat treat. As expected, there have been no cracks or flaws reported in either the nozzles or safe ends.

From the review of the materials used to construct the vessel and the results of the material surveillance program, it does not appear that this vessel has any low upper shelf problem. However, it was noted that some pertinent information on the weld material is missing. This item will be reviewed again when the staff receives this information.

Although no low temperature overpressurization incidents have been reported on San Onofre, NRC requested Southern California Edison Company take steps including design modifications, necessary to preclude exceeding the limits of Appendix G, 10 CFR Part 50, during inadvertent pressure transients (Ref. 14). By letters dated October 12, 1977, May 2, 1977, and March 7, 1978, the licensee submitted to NRC plant-specific analyses in support of their proposed reactor vessel low-temperature overpressure protection system (Ref. 15). Low temperature reactor vessel overpressure protection is provided by two pressurizer power-operated relief valves (PORVs). The OPS pressure setpoint is 522 psig. The system will be manually enabled during cooldown when the system pressure is less than the OPS setpoint and greater than 400 psig. Warning alarms have been installed to ensure that the system will be enabled. Pressure transients due to all postulated mass or heat additions (with the exception of inadvertent emergency core cooling system actuation) will be terminated below Appendix G limits by automatic operation of these valves. The licensee has proposed operating procedures that would decrease the probability of this inadvertent mass addition. The NRC staff is currently reviewing Southern California Edison Company's proposals.

Proposed changes to San Onofre's Technical Specifications have been submitted and are also under review (Ref. 16). The staff will require that the Technical Specifications include adequate provisions for testing and operating the PORVs and operating procedures to prevent the mass addition from inadvertent operation of the emergency core cooling system.

Conclusions and Recommendations

The San Onofre Unit 1 reactor vessel was designed to ASME Code Section VIII. However, the requirements of this Code were supplemented by the requirements of Nuclear Code cases, the Navy Code, and Westinghouse purchase specifications so that the design and quality control measures utilized were essentially in accordance with the rules of ASME Code Section III. Therefore, the initial integrity of the reactor vessel is considered acceptable. The primary stresses in the vessel beltline region are low, approximately 70% of those permitted by Section III. These low stresses, along with the use of materials with adequate fracture toughness, provide assurance that brittle fracture will not occur. Inservice inspections, conforming to the main requirements of ASME Code Section XI, have been performed on the reactor vessel since 1970. Welds in the vessel beltline region were ultrasonically examined in 1976 and no reportable indications were found. Only very minor indications have been detected on other vessel components. The staff will require that inservice inspections be continued in accordance with Section XI rules. The reactor vessel is currently operating with pressure-temperature operating limits that are in conformance with Appendix G, 10 CFR Part 50. The staff will continue to review and update these limits to account for additional radiation damage on the vessel materials. The amount of radiation damage will be predicted from the test results obtained on the vessels material surveillance specimens. The material surveillance program for San Onofre was reviewed and is considered to be very good. The combination of inservice inspections, conservative pressure-temperature operating limits, low vessel stresses, and the use of materials having acceptable fracture toughness properties provides that the vessel integrity will be maintained at acceptable levels throughout service life. The generic safety items applicable to San Onofre (low upper shelf toughness and overpressurization protection) have been successfully resolved and will not adversely affect the integrity of the vessel.

The following recommendation is made:

1. The licensee should attempt to get detailed information on the vessel and surveillance weld materials, such as the type and heat of weld wire and flux used, the chemical composition, and the results of any Charpy tests conducted on these materials.

References (Appendix D)

1. San Onofre FSAR, Section 2.5.*

2. Letter, Southern California Edison Company (K. Baskin) to NRR dated March 28, 1975.*

*Available in NRC PDR for inspection and copying for a fee

3. Westinghouse Reactor Vessel Equipment Specification E-569259, SCE Project, January 23, 1963.

4. Letter, Southern California Edison Company (K. Baskin) to Director, NRR, dated November 10, 1977.*

5. Southern California Edison Company Report, "Analysis of First Surveillance Capsule from San Onofre," dated July 1971.*

6. Southern California Edison Company Report, San Onofre Unit 1 Analysis of Second Surveillance Material Capsule, July 1972.*

7. D. J. Lege, et al., "Heatup and Cooldown Limits for Normal Operation for San Onofre," May 1, 1973.**

8. Letter, Southern California Edison Company, to Director, NRR, dated June 15, 1973.*

9. Letter, NRC (D. Skovholt) to Southern California Edison Company dated April 12, 1974.*

10. Southern California Edison Company Inservice Inspection Reports on San Onofre for 1970, 1973, 1975 and 1976.*

11. Letter, Southern California Edison Company (J. Head) to Director, NRR, dated November 21, 1978.*

12. Letter, Southern California Edison Company (K. Baskin) to Director, NRR, dated September 28, 1977.*

13. Letter, NRC (D. Ziemann) to Southern California Edison Company dated September 26, 1979.*

14. Letter, NRC (A. Schwencer) to Southern California Edison Company dated August 11, 1976.*

15. Letters, Southern California Edison Company (K. Baskin) to Director, NRR, dated May 2, 1977, October 12, 1977 and March 7, 1978.*

16. Letter, Southern California Edison Company (J. Moore) to Director, NRR, dated August 29, 1977.*

*Available in NRC PDR for inspection and copying for a fee.
**Available in source file for USNRC Report NUREG-0569.

APPENDIX E

HADDAM NECK PLANT
(CONNECTICUT YANKEE)
REACTOR VESSEL

Design

The nuclear steam supply system, consisting of four nearly identical heat transfer loops, was designed by Westinghouse. The reactor vessel was designed and fabricated by Combustion Engineering in accordance with ASME Code Section VIII, ASME Nuclear Code Cases 1270N and 1273N, and Westinghouse specifications (Ref. 1). In addition to the minimum Code requirements, the following quality enhancing factors were employed (Ref. 2):

1. The vessel stress analysis included analysis of thermal transient and fatigue effects. The analysis was performed in accordance with the method used in the Naval Reactors Program and given in PB-151987, "Tentative Structural Design Basis for Reactor Pressure Vessels and Directly Associated Components." The vessel stress analysis performed to the procedures outlined in this document, together with the Code and Code case requirements, is essentially equivalent to that required by ASME Section III for Class 1 vessels.

2. The vessel was constructed of SA-302, Grade B, plate and SA-336 forging material. These materials were Charpy V-notch and drop weight impact tested and are essentially equivalent to the SA-533, Grade B, Class 1, and SA-508, Class 2, materials being used today.

3. All plate and forging material forming the pressure boundary of the vessel was ultrasonically inspected.

4. Ferritic material surfaces were magnetic particle inspected prior to being covered with weld deposit.

5. All stainless steel cladding was dye penetrant inspected after final stress relief.

6. All unclad ferritic steel surfaces were magnetic particle inspected after final stress relief.

7. It was required that permanent records be kept of all fabrication inspection data.

A tabulation of fabrication in-process quality control inspections performed for the Haddam Neck reactor vessel, compared to those required by ASME Section VIII and Section III, is included in Table E-1.

TABLE E-1

INSPECTION AND ACCEPTANCE STANDARDS FOR HADDAM NECK

	ASME Section III		Connecticut Yankee		ASME Section VIII	
Materials	Exam.*	Extent	Exam.*	Extent	Exam.*	Extent
Plates	U.T.	100% Volume	U.T.	100% Volume	N.S.	-
Forgings	U.T.	100% Volume	U.T.	100% Volume	N.S.	-
Fabrication						
Weld Grooves	M.T. or P.T.	100% Surface	M.T.	100% Surface	Visual	N.S.
Weld Joints	M.T. or P.T.	100% Surface	M.T. or P.T.	100% Surface	-	-
Shell and Head	R.T.	100% Volume	R.T.	100% Volume	R.T.	100% Volume
Nozzle Welds	R.T.	100% Volume	R.T.	100% Volume	R.T.	100% Volume
	M.T. or P.T.	100% Surface	M.T. or P.T.	100% Surface	N.S.	-
Partial Penetration Welds						
Progressive	M.T. or P.T.	100% Surface	M.T. or P.T.	100% Surface	Visual	N.S.
Final Surface	M.T. or P.T.	100% Surface	M.T. or P.T.	100% Surface	Visual	N.S.

*Examination Notations
R.T. - Radiography
U.T. - Ultrasonic Examination
M.T. - Magnetic Particle Examination
P.T. - Liquid Penetrant Examination
N.S. - Not Specified

It is interesting to note that the Haddam Neck reactor vessel was reevaluated in accordance with Section III rules and found to be essentially in compliance with these rules (Ref. 2). In some areas, the design criteria even exceeded the Section III standards. (Refer to Attachment 1 of Reference 2.)

In the beltline region, the ID of the vessel shell is 154-5/16 inches and its minimum thickness is 10-5/8 inches. The vessel is clad with weld deposited 304 stainless steel to a minimum thickness of 5/32 inch. The design and operating pressures are 2485 psig and 2000 psig, respectively. The date of commercial operation is January 1, 1968. As of January 1, 1979, the plant has operated for about 8½ EFPY.

Materials

The beltline region of the reactor vessel consists of three shells: the lower, the intermediate and the nozzle shell (Ref. 3). Each of these shells is made from three SA-302, Grade B, plates welded together with vertical seams. The shells are welded together by circumferential welds. All welds were made by the submerged arc welding process using RACO3 weld wire and ARCOSB5 flux. The post-weld heat treatment was 1150°F ± 25°F for 40 hours followed by a furnace cool. Based on radiation levels (no copper contents are reported for weld materials),* the most probable limiting welds are the nozzle shell to intermediate shell circumferential weld (weld number 2-373) and two of the intermediate shell vertical seam welds (weld numbers 6-373B and C). The maximum neutron fluence predicted at end-of-life for these welds at the vessel ID is 3.5×10^{19} n/cm^2. The initial or unirradiated RT_{NDT} value of these welds is estimated to be 0°F in accordance with Branch Technical Position MTEB 5-2 (Ref. 4). The Charpy energy of these weld metals at a test temperature of 10°F varied from 50 to 59 ft-lbs. These values are above average for weld material.

The limiting plate materials are the three intermediate plates (plate numbers W9807-2, 4 and 7). Chemical analysis of these plates shows a maximum of 0.12% copper and 0.013% phosphorus. These values are relatively low and should provide a good resistance against radiation damage. The maximum fluence on the plate ID at end-of-life is predicted to be 4.2×10^{19} n/cm^2. For plate materials subjected to this fluence the upper-shelf energy, estimated in the weak direction by the methods in Reference 4, varies from 82 to 92 ft-lbs. The highest RT_{NDT} of these materials is 10°F. These values are above average for this material.

Material Surveillance Program

The material surveillance program for the Haddam Neck vessel is described in WCAP-7036 (Ref. 5). The program is based on ASTM E-185-66. It consists of eight materials test capsules which were located in the Haddam Neck reactor between the thermal shield and the vessel wall positioned opposite the center of the core. The test capsules are contained in baskets attached to the thermal shield. There are two capsule types, Type I and Type II. Only the Type II capsules contain weld metal specimens and there are three capsules of this type in the program.

*However, the copper content of the surveillance weld metal is 0.22%.

Each Type I capsule contains 32 Charpy V-notch specimens; 8 Charpy specimens are machined from each of the three Haddam Neck vessel plates; the remaining 8 Charpy specimens are machined from correlation monitor material. In addition, each Type I capsule contains 3 tensile specimens (1 specimen from each of the three vessel intermediate plates) and 6 WOL specimens (2 specimens from each of the three intermediate Haddam Neck plates). Dosimeters of copper, Al-Co and Cd shielded Al-Co 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: 8 specimens machined from one of the three Haddam Neck plates, 8 specimens of weld metal, 8 specimens of HAZ metal, and the remaining 8 specimens of correlation monitor material. In addition, each Type II capsule contains 4 tensile specimens and 4 WOL specimens; 2 tensile specimens and 2 WOL specimens from one of the three Haddam Neck plates and the weld metal. These capsules also contain dosimeters, and U^{238}, Np^{237}, and Co-Al and Ni wires. The surveillance weld metal is made from the same type and heat of weld wire and flux as the vertical seam welds in the vessel beltline region. Although not part of the actual surveillance program, Westinghouse has sponsored radiation damage tests on weld material identical to the intermediate to lower shell circumferential seam.

Since the program was initiated prior to the issuance of Appendix H, 10 CFR Part 50, it does not comply with all the Appendix H requirements. The main areas of noncompliance are: (1) 8 Charpy specimens in lieu of 12 required, (2) Charpy specimens of base material oriented in the strong direction, and (3) some capsules do not contain weld samples. None of these deficiencies are considered serious enough to limit the effectiveness of the program to monitor the radiation damage on the Haddam Neck vessel materials. We recommend that care be taken in selecting Charpy test temperatures to ensure that the upper-shelf is obtained by at least three points.

To date, two capsules (A and F) have been removed and tested. Capsule A specimens received an average fluence of 2.07×10^{18} n/cm^2 (Ref. 6). This capsule contained weld samples that showed a larger degree of radiation damage than the base materials. The upper-shelf energy of weld metal specimens dropped from 105 to about 75 ft-lb and RT_{NDT} increased by about 90°F. Capsule F received an average fluence of 4.04×10^{18} n/cm^2 (Ref. 7). This capsule contained no weld metal specimens. Westinghouse reviewed the results of the tests on Capsules A and F. They recalculated the fluence values as follows: 2.8×10^{18} n/cm^2 for Capsule A and 4.4×10^{18} n/cm^2 for Capsule F (Ref. 8). (From our review of the dosimeter data we conclude that the Westinghouse calculations are correct.) A third capsule was recently removed from Haddam Neck and has been tested. The final report on these tests has not been submitted to NRC as yet. However, preliminary information indicates that the fluence on this capsule was 1.8×10^{19} n/cm^2 This capsule also contained no weld metal specimens.

Pressure-Temperature Operating Limits

On April 26, 1973, Connecticut Yankee Atomic Power Company submitted a proposed change to the Haddam Neck Technical Specifications regarding pressure-temperature operating limits (Ref. 9). These limits are based on data from the two capsules removed from the vessel. Operating limit curves were submitted for operation through 4.8, 7.8, 14 and 30 EFPY. The limiting material is weld metal having an initial RT_{NDT} of 0°F. At 14 EPFY the fluence of a 1/4 T location is predicted to be about 1.0×10^{19} n/cm^2. By extrapolating the surveillance data in accordance with Regulatory Guide 1.99, Revision 1, RT_{NDT} will be 200°F at 14 EFPY. We conclude that these operating limits

are in accordance with Appendix G, 10 CFR Part 50. The Commission issued the requested changes to the Technical Specifications including the Safety Evaluation as License Amendment No. 25 on March 19, 1974 (Ref. 10). Haddam Neck is currently operating on the limits to 14 EFPY. Since these limits are based on extrapolating the surveillance data, the NRC staff will carefully rereview these operating limits and the 30 EFPY limits when new surveillance data at higher fluences becomes available.

Inservice Inspection Program

Although the ASME Code, Section XI, was nonexistent, a preoperational ultrasonic examination of the reactor vessel was performed by Southwest Research Institute (Ref 2). Acceptance standards for this inspection were from the ASME Code, Section III, 1965 Edition. It should be noted that, since there were no applicable inservice inspection codes in 1966 and the selection of components to be inspected was based primarily on the location of highly stressed areas as well as failure histories of similar nonnuclear components, all areas now requiring inspection under Section XI of the ASME Boiler and Pressure Vessel Code were not inspected.

Inservice inspections were performed by Southwest Research Institute in 1970 on some components of the reactor pressure vessel (Ref. 11). As part of the inservice examination of the Haddam Neck Nuclear Reactor Coolant System, Southwest Research Institute was asked by the licensee to pay special attention to the furnace-sensitized stainless steel safe ends which were made during the fabrication of the plant. The purpose of these examinations was to ensure that there was no evidence of stress corrosion cracking similar to that which had been observed in other nuclear power stations. No defect indications beyond acceptable ASME Boiler and Pressure Vessel Code standards were observed in the May 1970 inservice examinations.

Inservice inspections have also been performed by Westinghouse in 1971, 1973, 1975, 1976 and 1977 (Ref. 12). These examinations include portions of all welds that require examination except the beltline weld (Category B-A). This weld is scheduled to be inspected in 1980, which will be the end of the first 10-year inspection interval. The examinations conducted so far have detected no reportable indications in the reactor vessel.

To comply with the requirements of 10 CFR 50.55a(g), Connecticut Yankee Atomic Power Company, on June 29, 1977, submitted proposed revisions to the Haddam Neck Technical Specifications to implement an updated inservice inspection program (Ref. 13). This program is based on ASME Code, Section XI, 1974 Edition and Addenda through Summer 1975. Although the commercial operation date for Haddam Neck is January 1968, the existing inservice inspection program and first 10-year inspection interval started in April 1970 with the origin of the first drafts of ASME Section XI. As brought forth in Connecticut Yankee Atomic Power Company's (D. C. Switzer) letter to the USNRC (R. Purple) of May 26, 1976, an interval change took place to establish the commercial operation date as the ISI interval reference date per 10 CFR 50.55a(g) and to bring the Class 1 reactor coolant pressure boundary portion of the program into accord with scheduling of new requirements for Class 2 and 3 examinations. The Class 1 examination schedule will remain basically unchanged to allow the reactor vessel inspections to be conducted prior to the end of the existing interval. The new inspection program began on January 1, 1978 with the first inspection period of the second inspection interval. The inspection program was revised slightly by CYAPCO letter dated May 26, 1978 and the inservice testing program was revised by letter dated June 29, 1979. All examinations

will be conducted in accordance with Section XI requirements except for nozzle to vessel welds. CYAPCO has requested that a surface examination be substituted for volumetric examinations as the reinforcing collar on the nozzle precludes volumetric examination. The staff is expected to complete its review of this program during the fall 1979.

Generic Safety Items

The materials of the Haddam Neck reactor vessel were reviewed and it was concluded that at the end-of-life they would all have a Charpy upper-shelf energy above 50 ft-lbs. Therefore, we do not anticipate any low upper-shelf problem for Haddam Neck.

The Haddam Neck reactor vessel primary coolant nozzle safe-ends were made of wrought stainless steel attached prior to the final post-weld heat treatment on the vessel. Since this procedure sensitized the safe-end material, particular attention was paid to these safe-ends in the early inservice inspections (Ref. 11). Only one indication was found, and it had the appearance of an unfused area of a weld. It was removed by light grinding. No indications of stress corrosion cracking have been found to date.

Although no low temperature, overpressurization transients on the Haddam Neck reactor have been reported (Ref. 14), NRC requested on August 11, 1976 that efforts begin to design and install plant systems to mitigate the consequences of pressure transients at low temperatures (Ref. 15). It was also requested that operating procedures be examined and administrative changes be made to guard against initiating overpressure events. Connecticut Yankee Atomic Power Company responded to this request with a plant specific analysis (Ref. 16) and other information to support their proposed overpressure protection system (Ref. 17). Low temperature reactor vessel pressure protection is provided by two newly installed ASME Code safety valves (SV) mounted on the 3-inch (OD) line from the pressurizer steam space to the two power-operated relief valves (PORVs). Each SV has two motor-operated isolation valves (MOVs) isolating the SV from the pressurizer during normal operation. Whenever the RCS temperature is below 340°F, all four MOVs are opened and the two SVs provide overpressure protection. Each SV is set to open at 380 psig. A pressure transient caused by a mass addition (charging pump) or heat addition (decay heat, pressurizer heaters, or starting a reactor coolant pump with a hot steam generator) will be terminated below the Appendix G, 10 CFR Part 50, limits by automatic operation of the SVs. The Haddam Neck HPSIP is an extremely high pressure, large capacity pump. Based on the following factors, the licensee demonstrated that the HPSIP should be excluded as a design basis for the overpressure protection system.

1. Probability of an HPSIP mass addition is very low.

2. The cost to totally mitigate the HPSIP mass input is about $600K.

3. With the SVs, a HPSIP mass input would result in pressures only slightly above Appendix G limits.

The Haddam Neck OPS was fully installed and tested during the fall 1977 refueling outage. The Technical Specifications have been revised to include testing and administrative procedures recommended by the staff. These procedures, including the related Safety Evaluation, were incorporated into the Technical Specifications as part of Amendment No. 33.

Conclusion and Recommendations

The Haddam Neck reactor vessel was designed to ASME Code Section VIII. However, the requirements of this Code were supplemented by the requirements of Nuclear Code cases, the Navy Code and purchase specifications so that the design and quality control measures utilized were essentially in accordance with the rules of ASME Code Section III. Therefore, the initial integrity of the vessel is considered acceptable. Primary stresses in the vessel beltline region are low, approximately 3/4 of those permitted by Section III. These low stresses along with the use of material with adequate fracture toughness provide assurance that brittle fracture will not occur. Also, inservice inspections in accordance with ASME Code Section XI have been conducted on the reactor vessel since 1970. These inspections to date have revealed no indications that exceed Code allowable limits. Inservice inspections will continue throughout the vessel's service life in accordance with Section XI rules. The reactor vessel is currently operating with pressure temperature operating limits that are in accordance with Appendix G, 10 CFR Part 50. The staff will continue to review and update these limits to account for radiation damage on vessel materials. The degree of radiation damage will be determined from the results of the vessel's material surveillance program. The surveillance program for Haddam Neck has been reviewed and is considered acceptable. The combination of these inservice inspections, conservative operating limits, low vessel stresses and the use of materials having adequate fracture toughness properties provides assurance that the vessel integrity will be maintained at acceptable levels throughout service life. The generic safety items applicable to Haddam Neck (low upper-shelf energy and overpressurization protection) have been successfully resolved and will not adversely affect the vessel integrity.

The following recommendations are made:

1. The last two material surveillance capsules removed from Haddam Neck contained no weld metal samples. Therefore, it is recommended that another capsule be removed in the next several years. This capsule should contain weld metal specimens.

2. The present pressure-temperature operating limits are based on the extrapolation of data obtained from the material surveillance program. Since a capsule subjected to relatively high fluences has recently been removed from the vessel, we should have in the near future a better data base to estimate the amount of radiation damage. Therefore, the staff should review again the pressure-temperature operating limits when the test results on the recently removed capsule become available. If the results of these tests are not received by NRC by the fall of 1979, the licensee should be requested to submit them.

References (Appendix E)

1. Haddam Neck FDSA, Section V.**

2. Letter, Connecticut Yankee Atomic Power Company (D. Switzer) to R. Purple, NRC, dated March 20, 1975, with attached Reactor Vessel Equipment Specification 675-194 dated August 27, 1963.**

*Available at the National Technical Information Service, Springfield, VA 22161.
**Available in NRC PDR for inspection and copying for a fee.

3. Letter, Connecticut Yankee Atomic Power Company (D. Switzer) to A. Schwencer, NRC, dated November 4, 1977.**

4. NRC Standard Review Plan, Section 5.3.2.*

5. S. Yanichko, WCAP-7036, April 1967.**

6. D. Ireland and V. Scotti, Battelle Memorial Institute Report, October 30, 1970.**

7 J. Perrin, J. Sheckherd and V. Scotti, Battelle Columbus Laboratory Report, March 30, 1972.**

8. T. R. Mager, et al., WCAP-8121, April 15, 1973.**

9. Letter, Connecticut Yankee Atomic Power Company (D. Switzer) to Assistant Director for Operating Reactors dated April 26, 1973.**

10. NRC letter, D. Skovholt to Connecticut Yankee Atomic Power Company dated March 19, 1974.**

11. H. Hendricks and J. Porter, SwRI Report 17-2774, September 17, 1970.**

12. Westinghouse ISI Reports dated January 30, 1974, January 30, 1974, July 30, 1975, August 6, 1976 and January 12, 1978.**

13. Letter, Connecticut Yankee Atomic Power Company (D. Switzer) to A. Schwencer dated June 29, 1977.**

14. NRC Report, NUREG-0138, November 1976.*

15. NRC letter (A. Schwencer) to Connecticut Yankee Atomic Power Company dated August 11, 1976.**

16. Letters, Connecticut Yankee Atomic Power Company (D. Switzer) to Director, NRR, dated September 3, 1976, September 7, 1977 and March 6, 1978.**

17. Letters, Connecticut Yankee Atomic Power Company to Director, NRR, dated October 15, 1976, March 3, 1977, April 26, 1977, June 1, 1976 and January 30, 1978.**

*Available at the National Technical Information Service, Springfield, VA 22161.
**Available in NRC PDR for inspection and copying for a fee.

APPENDIX F

OYSTER CREEK NUCLEAR GENERATING STATION
REACTOR VESSEL

Design

The Oyster Creek nuclear steam supply system was designed by General Electric. It is a BWR/2 model having five recirculation loops. The reactor vessel was designed and fabricated by Combustion Engineering in accordance with the rules of ASME Code, Sections I and VIII, and Nuclear Code Cases 1270N and 1273N (Ref. 1). In addition to the above requirements, General Electric purchase specifications required the following quality enhancing factors (Ref. 2).

1. The vessel stress analysis included analysis of thermal transient and fatigue effects. The method used was largely based upon the method of analysis developed for naval reactors. The basic method is given in the Navy code PB-151987, "Tentative Structural Design Bases for Reactor Pressure Vessels and Directly Associated Components." The methods outlined in this document, slightly refined by GE, were included in the vessel purchase specification and the analysis performed in accordance with these requirements is essentially equivalent to that required by ASME Section III for Class 1 vessels.

2. The vessel was constructed of SA-302, Grade B plate and SA-336 forging material. These materials were Charpy V-notch and/or drop-weight-impact tested. The maximum original nil ductility transition temperature for vessel shell material opposite the core region was held to a maximum of plus 10°F and for material elsewhere in the vessel pressure boundary plus 40°F.

3. All plate and forging material forming the pressure boundary was ultrasonically inspected.

4. The surfaces of ferritic forgings and the weld grooves for ferritic materials were magnetic particle inspected.

5. All stainless steel cladding was ultrasonically inspected for bonding to the base metal and its surfaces were dye penetrant inspected.

6. The surfaces of completed pressure boundary welds were magnetic particle or liquid penetrant inspected.

A tabulation of material acceptance inspections for the Oyster Creek Unit 1 reactor vessel as compared to those required by ASME Section I and Section III is included in Table F-1.

TABLE F-1

MATERIAL AND ACCEPTANCE STANDARDS FOR OYSTER CREEK

	ASME Section III		Oyster Creek-1		ASME Section I	
Materials	Exam.*	Extent	Exam.*	Extent	Exam.*	Extent
Plates	U.T.	100% Volume	U.T.	100% Volume	N.S.	-
Forgings	U.T.	100% Volume	U.T.	100% Volume	N.S.	-
Fabrication						
Weld Grooves	M.T. or P.T.	100% Surface	M.T.	100% Surface	Visual	N.S.
Weld Joints	M.T. or P.T.	100% Surface	M.T.	100% Surface	-	-
Shell and Head	R.T.	100% Volume	R.T.	100% Volume	R.T.	100% Volume
Welds	M.T. or P.T.	100% Surface	M.T.	100% Surface	-	-
Nozzle	R.T.	100% Volume	R.T.	100% Volume	R.T.	100% Volume
Partial Penetration Welds						
Progressive	M.T. or P.T.	100% Surface	M.T. or P.T.	100% Surface*	Visual	N.S.
Final Surface	M.T. or P.T.	100% Surface	M.T. or P.T.	100% Surface	Visual	N.S.

*Examination Notations

R.T. - Radiography
U.T. - Ultrasonic Examination
M.T. - Magnetic Particle Examination
P.T. - Liquid Penetrant Examination
N.S. - Not Specified

In the beltline region, the reactor vessel ID is 213 inches and the vessel wall thickness is 7-1/8 inches. The inside of the vessel is clad with 304 stainless steel to a nominal thickness of 7/32 inch with weld-deposited E-308 electrode. The vessel operating pressure is 1000 psig and its design pressure is 1250 psig. At the end of life, the maximum fluence on the vessel wall is estimated to be 3.3×10^{18} n/cm^2. The date of commercial operation is December 1969. As of January 1, 1979, Oyster Creek has operated for about 6-1/2 EFPY.

Materials

The reactor vessel is comprised of a top head, a bottom head, and a cylindrical section. The top head is bolted to the flange of the cylindrical section. A double O-ring type seal is provided so that the area between the seals can be monitored for leakage. The cylindrical section is made from the shell flange and four shell courses. These courses are joined together by circumferential seam welds. Each shell course is made from three plates joined by longitudinal weld seams. The plates were purchased from Lukens Steel. The plate material is SA 302, Grade B steel modified by the addition of 0.3 to 0.7% nickel 3 (Ref. 3). This nickel addition makes this material identical to SA 533, Grade B, Class 1 steel. Nickel is added to increase the material's hardenability. A chemical analysis was performed on the plate materials but copper content was not reported. The maximum phosphorus content of the plates in beltline region was 0.019%.

The vessel welds were made by the automatic submerged metal arc process using RACO-3 weld wire and ARCOS B5 flux. A chemical analysis was not performed on actual weld metal. However, chemical analysis examinations were performed at Combustion Engineering on similar welds. Results indicate a maximum copper content of about 0.27% (Ref. 3). A chemical analysis was performed on each heat of weld wire used in the vessel fabrication. Again the copper content was not reported but the maximum phosphorus content of beltline weld wire was reported as 0.015%.

General Electric purchase specifications required all vessel materials, including weld metal samples from the core region, to be impact tested by either dropweight or Charpy methods. When the Charpy method was used for beltline materials, a complete Charpy curve was required (Ref. 2). All materials in the beltline region were required to have an NDT of 10°F or less and materials outside the beltline region were required to have an NDT of 40°F or less. The Charpy energy curves for all plate materials were reviewed. The average upper shelf energy for plate materials in the beltline region was over 100 ft-lbs for material oriented in the longitudinal direction. This is considered acceptable. The licensee was not able to find any impact test data on weld metal (Ref. 3).

The post weld heat treatment was at 1130°F ± 25°F, with a hold time of one hour per inch of thickness, and a furnace cool to 600°F.

Material Surveillance Program

The material surveillance program for Oyster Creek is based on ASTM Standard Practice E 185-70 (Ref. 4). The program originally consisted of three surveillance capsules containing specimens from vessel plate, weld and HAZ materials (Ref. 4). According to General electric purchase specifications, the surveillance plate material was to be machined from a leftover piece of a beltline plate (Ref. 2). The surveillance weld metal samples were to be taken from a weld that simulated a vessel longitudinal weld on plate material left over from beltline plates. However, we have no information on which beltline plates were used or the heats of flux and weld wire used (Ref. 3).

The number 1 capsule contains 12 Charpy specimens from each of plate, weld and HAZ materials. The number 2 capsule has nine specimens from each type of material and the number 3 capsule contains eight specimens from each material. The number 2 capsule also has nine Charpy specimens from a standard reference material. Each capsule also contains a minimum of two tensile specimens from each type of material. All specimens were machined from material at the 1/4 T location. Specimens from plate material were orientated in the longitudinal, or strong, direction.

Each capsule also contains copper, iron and nickel flux wires that will be used to measure the neutron flux and fluence. A special flux monitor capsule containing three flux wires from copper, iron and nickle was attached to the number 1 surveillance capsule. This capsule was designed to be removed independently of the surveillance capsule.

Since the Oyster Creek material surveillance program was planned prior to the initial issuance of Appendix H, 10 CFR 50, it does not meet all the requirements of this appendix. The main areas where it does not meet Appendix H requirements are:

1. Insufficient number of capsules.

2. Insufficient number of Charpy specimens in the capsules.

3. Charpy specimens orientated in the longitudinal direction instead of the transverse direction.

Originally the surveillance program consisted of three capsules which is in accordance with Appendix H requirements. However, the number 1 capsule was removed from the vessel after about two years of operation. This early withdrawal resulted from a design error which made the independent withdrawal of the flux capsule impossible. The flux capsule was inadvertently tack-welded to the surveillance capsule. Thus, when the flux capsule was scheduled for removal, it was decided to remove both the flux capsule and the number 1 surveillance capsule (Ref. 5). Attempts to replace the surveillance capsule were futile. The number 1 capsule is presently located in the Oyster Creek spent fuel pool. Since the fluence levels on specimens in this capsule are very low, there are no plans to test these specimens. The staff agrees with this plan. However, should the need arise, these specimens could be irradiated in a test reactor where they would receive high levels of radiation in a relatively short period of time. Also, there are 24 Charpy specimens from each type of surveillance material stored at General Electric, San Jose. If needed these specimens could also be irradiated quickly in a test reactor. Therefore, the staff concludes that the low number of capsules does not significantly reduce the effectiveness of the Oyster Creek program. The staff also concludes that the insufficient number of Charpy specimens and the wrong orientation of specimens does not make the program unacceptable provided that care is taken in the selection of test temperatures so that the upper shelf energy is obtained. Therefore, we conclude that this surveillance program is acceptable.

To date neither of the two remaining material surveillance capsules has been removed from the reactor vessel and tested. The flux dosimeter capsule, removed from the vessel in the fall of 1971, was analyzed at Battelle-Columbus Laboratories (Ref. 6). From these flux wires it was determined that the maximum fluence on the vessel wall ID at the end of life will be 3.3×10^{18} n/cm^2.

The present Technical Specifications for Oyster Creek do not contain a surveillance capsule withdrawal schedule. The licensee has recommended that none of the capsules be withdrawn and tested because of the relatively low fluence values on the vessel wall (Ref. 7). The staff does not agree with this recommendation. We recommend that one capsule, preferably the number 2 capsule, should be removed at about 12 to 15 EFPY and tested. This should give results for radiation damage at fluences approximately equal to those expected on the vessel wall at end of life. The number 3 capsule should be designated as a standby capsule and will be removed only if needed.

Pressure-Temperature Operating Limits

The original pressure-temperature operating limits for Oyster Creek were based on a criteria using RT_{NDT} +60°F. These operating limits were reviewed in 1973 and 1974 as part of the review of the licensee's application for a Full Term Operating License (FTL). From this FTL review, it was concluded that these operating limits were not in accordance with the requirements of Appendix G, 10 CFR 50. Therefore, the licensee was requested to provide revised operating limits in accordance with Appendix G (Ref. 8). The licensee responded in Supplement No. 6, Addenda 3, to its FTL application with revised operating limits calculated in accordance with Appendix G, 10 CFR 50 (Ref. 9). The NRC technical staff reviewed these proposed operating limits and concluded that they were acceptable (Ref. 10). Then the licensee formally requested a change to the Technical Specifications for Oyster Creek, change No. 33, to incorporate these proposed limit curves into the specifications (Ref. 11). Shortly after this the FTL review was postponed pending redefinition of the scope of review for FTL applications. With this postponement the requested change was never granted and the proposed operating limits were never incorporated into the Technical Specifications. Therefore, the present Technical Specifications contain the original pressure-temperature operating limits that are not in accordance with Appendix G. However, the licensee notified the NRC staff that Oyster Creek is currently operating with the proposed limit curves. This is acceptable and not in violation of the Technical Specifications since the proposed limits are more conservative than those in the specifications. The NRC staff was presented with a second chance to update this specification when, in response to an NRC question regarding feedwater nozzle cracking, the licensee stated that revised operating limits in accordance with Appendix G had been submitted to NRC by letter dated January 9, 1975 (Ref. 12). However, the staff again took no action on the licensee's request.

On October 3, 1979, the licensee submitted updated pressure-temperature operating limits conforming to Appendix G. These limits are based on an end of life fluence on the vessel wall ID of 3.3 10^{18} n/cm^2. At the 1/4 T location, this value will attenuate to about 2 x 10^{18} n/cm^2. The limiting material is judged to be weld metal having a copper content of 0.27%. Since no material surveillance capsules have been tested to date, Regulatory Guide 1.99, Revision 1, is used to estimate the radiation damage (the increase in RT_{NDT}) on vessel materials. These revised limits were incorporated into the Technical Specifications by Amendment No. 42 dated October 16, 1979.

Inservice Inspection Program

The inservice inspection presently in the Technical Specifications is the original inspection program. It generally conforms to the rules of the 1971 Edition of ASME Code, Section XI. Note 3 of Table 4.3-1 of this specification requires that the licensee develop a revised inservice inspection program after the fourth year of operation. In accordance with this requirement, Jersey Central Power and Light submitted a revised inservice inspection program to NRC by letter dated

August 6, 1974 (Technical Specification Change Request No. 28) (Ref. 13). This revised program was based on the 1971 Edition of Section XI and Addenda through the Summer 1973. This revised program was never incorporated into the Oyster Creek Technical Specifications.

This proposed program indicates that most of the welds requiring examination can be inspected according to Section XI rules with the exception of Category B-A welds in the core region and some of the Category B-B vessel welds. It appears that only the upper six feet of the longitudinal vessel welds in the upper shell course can be volumetrically examined. However, the circumferential head-to-flange and flangeto-shell welds can be examined. Also most of the reactor nozzle and safe end welds can be examined according to code rules. Although this proposed inservice inspection program was never incorporated into the Technical Specifications, the licensee has used this program along with the program in the Technical Specifications as a guide for inservice inspections on Oyster Creek. Thus, to date, examinations have been performed on the reactor vessel that exceed the requirements of the Technical Specifications.

Inservice examinations have been performed on various components of the reactor vessel in 1971, 1972, 1973, 1974, 1975 and 1978 (Ref. 14). Examinations have been made on some welds of each inspection category except for the Category B-A welds in the beltline region. These examinations revealed some cracks in the vessel head cladding. Evaluation of these cracks indicated that they were intergranular stress corrosion cracks in the clad material that did not penetrate into the base material. The flawed areas were reexamined in the 1975 inspection and no crack growth was detected. The only other cracks or flaws found in these inspections were cracks in the feedwater nozzles. These are discussed in more detail in the Generic Safety Item section of this appendix.

A complete preservice examination was not performed on this vessel. However, extensive nondestructive preservice examinations were performed on certain vessel components, especially those suspected to have been made from sensitized stainless steel. These examinations were part of the "Reactor Vessel Repair Program." This program was initiated because a leak near a control rod drive housing was detected during a hydrostatic test on September 29, 1967 (Ref. 15). Examination of the flawed area indicated that the crack was caused by incomplete fusion in the weld. Further examination, by dye penetrant method, revealed a number of small cracks in stub tubes. Based on the results of the above examinations, the licensee decided to perform additional nondestructive examinations on other areas of the vessel (Ref. 16). Specifically, the following components were examined:

1. Ten recirculation system inlet and outlet nozzle transition welds

2. Nine recirculation system nozzle safe ends (inside and outside)

3. Approximately 40 instrumentation, head spray, vent and other nozzle transition welds

4. Approximately 20 internal structure bracket welds

5. Shroud support cone welds

6. Shroud support cone stainless steel upper flange.

These examinations revealed enough significant flaws so that a repair program was initiated. The three nozzles located on the vessel head were replaced with nonsensitized material. Stub tubes

were removed and repaired, then overlayed with 308L weld metal, and rewelded to the housing. All sensitized stainless steel safe ends, except for three on the vessel head that were replaced, were overlayed with 308L weld metal on both the ID and the OD. Following these repairs, components were examined by dye penetrant and ultrasonic methods (Ref. 17). Only several very small spot types of indications were detected in this examination. These indications were removed by grinding. The vessel repair program is considered to be acceptable and it is concluded that the initial integrity of the vessel was acceptable.

Oyster Creek is scheduled to submit a revised inservice inspection program to NRC in accordance with the provisions of 10 CFR 50.55a(g) in the latter part of 1979. Since the present Technical Specifications presently contain an outdated inspection program, the revised program should be used as a basis for any examinations following its submittal even though the staff has not reviewed it.

Generic Safety Items

The generic safety items reviewed for Oyster Creek are low upper shelf toughness, sensitized stainless steel safe ends, feedwater nozzle cracking and control rod drive return line nozzle cracking.

To date no surveillance capsules have been withdrawn from the Oyster Creek reactor vessel. However, because of the low fluences on the vessel beltline materials, the chemistry of these materials, and the mechanical properties of the unirradiated materials, it is concluded that the Oyster Creek reactor vessel materials will have acceptable upper shelf energy values throughout service life. This item will be reviewed again when the test results on irradiated surveillance specimens becomes available.

As originally constructed, the reactor vessel had a number of sensitized stainless steel safe ends. As part of the "Reactor Vessel Repair Program," steps were taken to correct this deficiency. Safe ends in the vessel head were replaced with nonsensitized material. Other safe ends were overlayed with 308L weld metal on both the OD and the ID. Prior to being overlayed, all flaw indications were removed by grinding. Since these repairs were made, no indications of flaws have been detected in these safe ends during subsequent inservice inspections. Therefore, it is concluded that the repairs were acceptable and that the sensitized stainless steel safe end problem has been successfully resolved.

The Oyster Creek feedwater nozzles were examined by Breda Thermomeccanica in 1976 using ultrasonic techniques. Only one indication was detected in the base material and that did not appear to be a stress corrosion crack (Ref. 18). These feedwater nozzles were again inspected in 1977. Ultrasonic examination methods were again used. However, the examination techniques and procedures were improved. Twelve cracks extending into the base material were found (Refs. 19, 20). The maximum penetration into base metal was about 1/4 inch. Following this inspection, the stainless steel cladding was removed and all flaws were removed by grinding. New feedwater spargers and nozzle thermal sleeves designed by MPR Associates were installed. These new spargers were made of carbon steel and Inconel 600. Both of these materials have performed satisfactorily in BWR service. The thermal sleeve sealing end incorporates a piston ring and the sleeve was installed with a 10 mil interference fit. The sparger assembly included two flow shrouds that were preloaded against the reactor vessel wall to prevent hot reactor water from entering the annulus between the sleeve and the nozzle. The staff has evaluated this new design and concludes that it is acceptable. However,

the staff will still require augmented inservice examinations to be conducted on these feedwater nozzles. The staff will monitor the results of these examinations for any signs of further cracking.

In 1977 the CRD RL nozzle thermal sleeve which was welded to the nozzle was removed and a dye penetrant examination was performed on the inside diameter of the nozzle (Ref. 20). No indication of cracking was found. The thermal sleeve was then reinstalled. The CRD RL nozzle has been in operation for more than six years and no cracking has been observed. By design the thermal sleeve protrudes into the vessel. This design prevents the flow of incoming cold water from contacting the vessel wall. Since there has been no indication to date of thermal fatigue cracking in the Oyster Creek CRD RL nozzle, the staff concludes that the present design is acceptable. However, augmented inspections will be required on this nozzle. Again, the staff will monitor the results of these examinations for any signs of cracking.

Conclusions and Recommendations

The Oyster Creek reactor vessel was designed to ASME Code, Sections I and VIII. However, the requirements of this code were supplemented by the requirements of nuclear code cases, the Navy Code and General Electric purchase specifications so that the design and quality control measures utilized were essentially in accordance with the rules of ASME Code, Section III. Therefore, the initial integrity of the vessel is considered acceptable. Primary stresses in the vessel beltline region are low, approximately 3/4 of those permitted by Section III. These low stresses along with the use of materials having adequate fracture toughness provide assurance that brittle fracture will not occur. Inservice inspections have been performed on the reactor vessel in accordance with ASME Code, Section XI rules since 1971. To date these examinations have detected no serious flaws except for those found in the feedwater nozzles. These cracks have been repaired, and redesigned feedwater spargers and thermal sleeves have been installed. Inservice examinations in accordance with Section XI rules will continue to be performed on vessel components throughout service life. The pressure-temperature operating limits in the Technical Specifications are in accordance with Appendix G, 10 CFR 50. The NRC staff will continue to review these operating limits and update them, as necessary, to account for radiation damage on the vessel materials. The amount of radiation damage will be determined from Regulatory Guide 1.99 until about 12 EFPY. At about 12 EFPY a surveillance capsule should be removed from the vessel and tested. The results of tests on this capsule will then be used to predict the amount of radiation damage. The material surveillance program for Oyster Creek was reviewed and is considered acceptable. The combination of inservice inspections, conservative pressure-temperature operating limits, low stresses in the vessel and the use of materials having adequate fracture toughness properties provides assurance that the vessel integrity will be maintained at acceptable levels throughout service life. The generic safety items applicable to Oyster Creek (low upper-shelf toughness, sensitized stainless steel safe ends, feedwater nozzle cracking and control rod drive return line nozzle cracking) have been successfully resolved for the purposes of this report and will not adversely affect the integrity of the vessel. NRC Technical Activity A-10 will complete the resolution of the feedwater and CRD RL nozzle issues.

The following recommendations are made:

1. The Technical Specifications presently do not contain a material surveillance capsule withdrawal schedule. There are two capsules presently in the reactor vessel. It is recommended that the number 2 capsule be removed at about 12 EFPY and tested. The number 3 capsule should be designated as a standby capsule.

2. In response to NRC request for information on the reactor vessel material surveillance program (NRC letter to Jersey Central Power and Light Co. dated May 20, 1977), the licensee responded with information in letter dated January 12, 1978. However, the response gave incomplete information regarding Question 5. This information should be requested again.

3. The Technical Specifications currently contain the original inservice inspection program. This program is far outdated. Therefore, an updated inservice inspection program should be incorporated into the Technical Specifications as soon as possible.

References, Appendix F

1. Oyster Creek Facility Description and Safety Analysis Report, Section IV (FDSAR).*

2. Amendment 16 to the FDSAR.*

3. Jersey Central Power and Light Company (JCPL) letter (I. Finfrock) to NRR dated January 12, 1978.*

4. Section 2.0 of Amendment 48 to the FDSAR.*

5. Supplement 7, Amendment 68 to Application for FTL dated January 9, 1975.*

6. Supplement 6, Addendum 3 to the Application for FTL dated April 8, 1974.*

7. Supplement 4 to the Application for FTL dated March 5, 1973.*

8. Letters NRC, ORB-1 (R. Schemel) to JCPL dated July 3, 1973 and August 7, 1973.

9. JCPL letter (I. Finfrock) to NRC (ORB-1) dated April 8, 1974.

10. NRC Internal Memo, Safety Evaluation from R. Maccary to K. Goller dated January 30, 1975.**

11. JCPL letter (I. Finfrock) to NRC (ORB-3) dated January 9, 1975.*

12. JCPL letter (I. Finfrock) to NRC (ORB-3) dated May 18, 1976.*

13. JCPL letter (I. Finfrock) to NRC (ORB-3) dated August 6, 1974.*

14. Oyster Creek Semiannual Operating Reports Numbers 5, 6, 8, 10 and 12.*

15. Reactor Vessel Repair Program Reports, Amendments 29, 35, 37, 40, 43 and 47 to the FDSAR.*

16. Amendment 35 to the FDSAR.*

*Available in NRC PDR for inspection and copying for a fee.
**Available in source file for USNRC Report NUREG-0569.

17. Amendments 43 and 47 to the FDSAR.*

18. JCPL letter (I. Finfrock) to NRC (ORB-3) dated May 14, 1976.*

19. Preliminary Report of Unusual Occurrence-PNO-77-101 dated June 13, 1977.*

20. JCPL letter (I. Finfrock) to NRR, dated September 22, 1978.*

*Available in NRC PDR for inspection and copying for a fee.

APPENDIX G

R. E. GINNA NUCLEAR POWER PLANT, UNIT 1
PRESSURE VESSEL

Design

The nuclear steam supply system of R. E. Ginna Nuclear Power Plant, Unit 1, consisting of two identical heat transfer loops, was designed by Westinghouse. The reactor vessel was designed and fabricated by Babcock and Wilcox Company in accordance with Westinghouse specifications and to the rules of ASME Code Section III, 1965 Edition (Ref. 1). The Westinghouse specifications required that many tests and examinations not required by Section III be conducted (Ref. 2). Some of the more important of these supplementary requirements were:

1. Full Charpy V-notch curves (upper and lower shelves included) for all vessel plate and forging material and vessel beltline weld materials. The supplier was to "aim" for obtaining material with a Charpy V-notch transition temperature of less than 10°F and an NDT temperature of less than 10°F from drop weight tests.

2. All plate material was examined by shear waves performed in two perpendicular directions.

3. All cladding was ultrasonically examined to verify the bond to the base metal and dye penetrant examined after the hydrostatic test.

4. Following the hydrostatic test, various high stressed regions of the vessel were ultrasonically examined, such as the coolant nozzle, middle to lower shell course weld, nozzle shell to middle shell course weld, flange to nozzle shell course weld, nozzle shell course and lower vessel flange taper.

In the beltline region the vessel ID is 132 inches and its wall thickness is 6.5 inches. The minimum clad thickness is 0.156 inch. The design and operating pressures are 2485 psig and 2235 psig, respectively. The date of commercial operation is July 1970. As of January 1, 1979, the plant has operated for about 6 EFPY.

Materials

The cylindrical section of the reactor vessel is comprised of three cylindrical forgings fabricated from SA 508, Class 2, material (Ref. 2). The top and bottom dome sections are made from SA 533, Grade A, plate material. The shell course, flanges and nozzles were made from SA 508, Class 2, forgings. The vessel materials opposite the core were purchased to a specified Charpy V-notch impact energy of 30 ft-lbs or greater at a corresponding nil ductility temperature (NDT) of 40°F or less. These materials were subsequently tested (drop weight) to determine their actual NDTs and to verify that they were less than 40°F (Ref. 1). The copper and phosphorus content of the beltline forgings is reported to be no greater than 0.07% and 0.012%, respectively. These values are low and should provide a good resistance against radiation damage.

There are two welds in the beltline region; i.e., the nozzle shell to intermediate shell (SA-1101) and the intermediate shell to lower shell (SA-847) (Ref. 3). Both are circumferential welds made by the submerged arc process using Mn-Mo-Ni weld wire and Linde 80 flux. It is known that this combination of weld wire and flux has produced some welds that have relatively low Charpy upper-shelf energy values. The post-weld heat treatment was 48 hours at 1100°F-1125°F followed by furnace cooling. The reported copper content of these welds is 0.21% and 0.20%, respectively.* The maximum predicted end-of-life fluence on the weld ID is 2.0 x 10^{18} n/cm^2 for SA-1101 and 3.7 x 10^{19} for SA-847. The initial or unirradiated RT_{NDT} of these welds is estimated to be 0°F in accordance with Branch Technical Position MTEB 5-2 (Ref. 4). Based on its exposure to radiation and its chemical composition, weld SA-847 is the limiting vessel material.

For forging materials in the beltline region, upper-shelf energies (estimated in the weak orientation) are above 90 ft-lbs. These values are considered to be excellent. No upper-shelf energy values were determined for weld metals. However, the upper-shelf energy of the surveillance weld metal is 80 ft-lbs. This value is also above average. Charpy energy at a test temperature of 10°F for the beltline weld materials varied from 45 to 67 ft-lbs.

Material Surveillance Program

The material surveillance program for Ginna is described in WCAP-7254 (Ref. 5). The program is based on ASTM E-185-66. It consists of six surveillance capsules positioned in the reactor vessel between the thermal shield and vessel wall. The vertical center of the capsules is opposite the vertical center of the core. The lead time for these capsules varies from 1.4 to 3.3. Each capsule contains tensile, Charpy V-notch and WOL specimens from the forgings (heats 125P666 and 125S255) and weld metal, and Charpy V-notch specimens from HAZ material and from an A302, Grade B, correlation monitor material furnished by U.S. Steel Corporation. All test specimens were machined from the 1/4 T location of the forgings. Test specimens represent material taken at least one forging thickness from the quenched end of the forging. All Charpy V-notch and tensile specimens were oriented with the longitudinal axis of the specimen parallel to the hoop direction of the forgings (strong direction). The WOL test specimens were machined with the simulated crack of the specimen perpendicular to the surfaces and the hoop direction of the forgings.

The data obtained on the correlation material provide us with a comparison of radiation damage in a commercial reactor vessel with a test reactor. These data also give us an indication of the accuracy of the neutron fluence calculations.

The surveillance weld metal is designated as SA-1036 (Ref. 3). The SA-1036 weld was made from the same weld wire as the SA-847 weld (the limiting reactor vessel weld). However, a different flux was used. Thus, the surveillance weld is not identical to the limiting vessel weld. However, we presently feel that the weld wire affects the properties of the weld more than the flux does. Therefore, we consider the surveillance weld to be representative of the vessel limiting weld and it should provide adequate information on radiation damage.

Capsules also contain dosimeter wires of copper, nickel and aluminum-cobalt, and cadmium shielded dosimeters of Np^{237} and U^{238}.

*In view of recent data on B&W welds, these values may be low. The chemistry of the surveillance weld should be rechecked when the next capsule is pulled and tested.

Since this program was conceived prior to the issuance of Appendix H, 10 CFR Part 50, it does not comply with all the requirements of Appendix H. The main areas of noncompliance are: (1) the lead factor of 3.3 for 2 capsules is slightly higher (10%) than the maximum value of 3 required by Appendix H,* (2) there are only 10 Charpy specimens in lieu of the required 12, and (3) Charpy specimens are oriented in the strong direction. None of these discrepancies is serious enough to limit the effectiveness of the program. We conclude that the Ginna material surveillance program will provide sufficient data for us to monitor the effects of neutron exposure on the mechanical properties of the reactor vessel core region materials throughout their service life.

To date, two capsules have been removed from Ginna and tested. Capsule V was removed during the second refueling shutdown after 2½ years of operation and tests were conducted at Southeast Research Laboratories (Ref. 6). Specimens received an average neutron fluence (E > 1 mev) of 4.9×10^{18} n/cm^2. The limiting vessel material is weld metal. This fluence resulted in an RT_{NDT} shift at 30 ft-lb for weld metal of 140°F. Base materials showed a maximum shift of 25°F. The upper-shelf energy of weld metal dropped from 80 to 55 ft-lb (based on only one data point).

Capsule R was removed after 4 years of operation and tested at Battelle Columbus Laboratory (Ref. 7). Specimens received an average fluence of 7.6×10^{18} n/cm^2. This fluence resulted in an RT_{NDT} increase of 170°F at 30 ft-lbs and a drop in upper-shelf energy to 51 ft-lbs for the weld metal. The base materials again showed very little radiation damage.

The radiation damage observed from these test results is very close to that predicted in Regulatory Guide 1.99, Revision 1.

A dosimetry analysis was made on flux monitors removed with the surveillance specimens. The original analysis made on Capsule V monitors was upgraded in the report on Capsule R (Ref. 7). The analysis showed an excellent comparison between calculated and measured neutron fluxes. Based on these analyses, the maximum end-of-life (32 EFPY) fluence on the vessel wall ID is 3.7×10^{19} n/cm^2.

The Technical Specifications require the next capsule to be withdrawn at the nearest refueling outage to 10 years of operation.

Pressure-Temperature Operating Limits

On March 10, 1975, Rochester Gas and Electric Corporation submitted an application to amend the Ginna Technical Specifications regarding pressure-temperature operating limits (Ref. 8). These limits are based on data obtained from tests on the two surveillance capsules removed from the reactor vessel. The operating limits were proposed for operation through 10.6 EFPY, at which time the maximum fluence on the vessel wall at 1/4 T location is estimated to be 7.4×10^{18} n/cm^2. Since this is less than the fluence on the second withdrawn capsule, we feel that the surveillance results provide an adequate basis for the proposed limits. We also concluded that the proposed operating limits are in accordance with Appendix G, 10 CFR Part 50. The Commission issued the requested changes to the Technical Specifications and a safety evaluation as part of License Amendment No. 12 on April 7, 1977 (Ref. 9).

*According to the analysis in Reference 7, the lead factor for these two capsules may be 2.5 instead of the 3.3 initially reported.

Inservice Inspection Program

In August 1969, Southwest Research Institute performed a preservice nondestructive examination on selected components, including the reactor vessel at Ginna, prior to the system's startup (Ref. 10). These examinations were performed to evaluate the overall structural integrity of the system and to provide baseline information for subsequent inservice examination comparisons. A manual ultrasonic technique was employed. The procedure and sensitivity requirements for the examination were derived from Southwest's experience and are in accordance with Section III, Appendix IX, of the ASME Code. Subsequent inservice examinations have been conducted by Southwest Research Institute in 1971, 1972, 1974, 1975, and 1978 (Ref. 11). The first inspection interval will be completed in 1979, at which time the beltline circumferential weld will be examined. Upon completion of the 1979 inspections the Ginna vessel will have been examined in accordance with Section XI requirements for the first inspection interval.* To date no unacceptable indications have been reported.

Rochester Gas and Electric Corporation submitted a proposed inservice inspection program for Ginna on February 27, 1977 in accordance with paragraph 50.55a(g) of 10 CFR Part 50 (Ref. 12). The program is for the third 40-month inspection period of the first inspection interval. It is based on ASME Code Section XI, 1974 Edition and Addenda through Summer 1975. All welds listed in Table IWB-2600, Section XI, under Reactor Vessel are scheduled to be examined according to Code rules. This program was reviewed by NRC in March 1977 and found acceptable. The Technical Specifications of Ginna were revised by License Amendment No. 13 to include this new inspection program (Ref. 13).

By letter dated July 2, 1979, Rochester Gas and Electric Corporation submitted an inspection program for the second inspection interval on Ginna. This proposed program is currently being reviewed by the NRC staff.

Generic Safety Items

The generic safety items for Ginna covered in this report are low upper-shelf toughness and overpressurization protection. Based on data from the two material surveillance capsules removed from Ginna, it now appears that this reactor vessel may have materials (weld metal) in the beltline region that will fall below 50 ft-lb after 11 to 13 EFPY when the fluence is approximately 8×10^{18} n/cm^2. However, it must again be pointed out that the surveillance weld metal is not identical to the limiting vessel weld. Currently, the staff is studying the effect of residual elements, weld wire and flux, and heat treatment on the irradiated mechanical properties of weld metals. This study should be sufficiently complete by the early 1980s to allow us to make an accurate determination of the minimum acceptable upper-shelf energy required for Ginna and the time when its limiting vessel material will reach this energy level. The low upper-shelf energy problem is also being evaluated under NRC Category A, Technical Activity A-11, "Reactor Vessel Materials Toughness" (Ref. 14). This activity is currently scheduled to be completed by the end of 1979. This activity

*An inservice inspection was performed on components of the Ginna reactor vessel in February-March 1979. During this inspection, the vessel beltline welds were examined and no indications exceeding Section XI allowable limits were found. However, an indication in the inlet nozzle N2B to vessel weld that exceeded Code allowable limits was detected. This indication was sub-surface and approximately 0.9 inches deep. The flaw is believed to be a slag inclusion. Teledyne performed a fracture mechanics analysis of the indication in accordance with Section XI criteria for acceptance by evaluation. It was determined the flaw is conditionally acceptable for continued operation without repair (reference Teledyne Report TR-3454-1, March 15, 1979). The NRC staff will closely watch the results of future examinations for any signs of flaw growth.

will provide us with a criterion to ensure that adequate safety margins are maintained for vessels having low upper-shelf energies.

In August 1976, NRC requested that Rochester Gas and Electric Corporation take steps, including design modifications, necessary to preclude exceeding the limits of Appendix G, 10 CFR Part 50, during inadvertent pressure transients (Ref. 15).

By letter dated July 29, 1977, Rochester Gas and Electric Corporation (RG&E) submitted to the NRC a plant-specific analysis in support of the proposed reactor vessel overpressure protection system (OPS) for the Ginna Nuclear Power Plant (Ref. 16). This information supplements other documentation submitted by RG&E over the preceding 12 months (Ref. 17). Low-temperature reactor vessel overpressure protection is provided by the two pressurizer power (air) operated relief valves (PORVs). The licensee modified the actuation circuitry of these valves to provide a low pressure setpoint of 435 psig. Whenever the RCS temperature is below 330°F, the low pressure setpoint is manually enabled using four keylock switches in the control room. A pressure transient caused by mass addition (HPSIP or charging pump) or heat addition (decay heat, pressurizer heaters or starting a reactor coolant pump with a hot steam generator) is terminated below the Appendix G limits by automatic operation of these valves. Most of the equipment needed for the OPS was installed during the 1978 refueling outage. Several items, such as seismic qualified valves, will be installed later as they are not currently available. The Technical Specifications have been revised to include testing and administrative procedures recommended by the staff. These specifications, including the related Safety Evaluation, were incorporated into the Technical Specifications by Amendment No. 26 (Ref. 18).

Conclusions and Recommendations

The Ginna reactor vessel was designed to the 1965 Edition of ASME Code, Section III. The requirements of this code were supplemented by the requirements of Westinghouse's specifications that required fracture toughness tests and nondestructive examinations above those required by the Code on the reactor vessel materials. To further verify the initial integrity of the reactor vessel, a complete preservice examination was conducted. From our review of the design criteria and the results of the preservice examinations, it is concluded that the initial integrity of the reactor vessel was acceptable. Inservice inspections in accordance with ASME Code Section XI have been conducted on the vessel since 1971. To date, these examinations have revealed no unacceptable indications. The effect of irradiation on the mechanical properties of the vessel beltline materials is monitored by Ginna's material surveillance program. This program has been reviewed by NRC and is considered acceptable. The reactor vessel is currently operating under conservative pressure-temperature operating limits that are in conformance with Appendix G, 10 CFR Part 50. The staff will continue to review and update these operating limits to account for additional radiation damage on vessel materials. The degree of this radiation damage will be determined from the results of tests on the material surveillance specimens. At present, the vessel materials have acceptable toughness properties. However, the future integrity of the Ginna reactor vessel may be limited by the toughness properties (upper-shelf Charpy energy) of the vessel beltline weld metal which is predicted to fall below 50 ft-lbs after 11 to 13 EFPY. Therefore, at this time, we conclude that the combination of inservice inspections, conservative operating limits and vessel materials having acceptable fracture toughness properties provides assurance that the vessel integrity is acceptable for operation through 11 EFPY. It is noted that another surveillance capsule is scheduled for removal from the vessel at 10 EFPY. It is recommended that the staff rereview this low upper-shelf energy problem

at about 10 EFPY when the data from the above capsule can be factored into the review. The other generic safety item applicable to Ginna, overpressurization protection, has been satisfactorily resolved. Therefore, the only item that affects the integrity of this vessel for operation past 11 EFPY is the vessel beltline welds that are predicted to have low upper-shelf toughness.

From this review the following recommendations are made:

1. The next surveillance capsule is scheduled for removal at 10 EFPY. It is recommended that this capsule be removed during the refueling outage just prior to 10 EFPY so the results will be available for the review of the vessel low upper-shelf problem.

2. In testing the weld metal Charpy specimens, particular attention should be devoted to obtaining an accurate upper-shelf energy value with a minimum of three data points. It is noted that in one of the previous test series that only one data point was really on the upper-shelf.

3. The staff should rereview the fracture toughness of the vessel beltline materials at about 10 EPFY and determine if these materials meet the requirements of Appendix G, 10 CFR Part 50.

4. A complete chemical analysis should be performed on a minimum of three Charpy specimens. The specimens may be from a capsule previously tested or from the next capsule removed from the reactor vessel.

References (Appendix G)

1. R. E. Ginna, *Technical Supplement Accompanying Application for FTL*, August 1972.**

2. Westinghouse Equipment Specification No. 676413, Rev. 1, October 28, 1966.***

3. Rochester Gas and Electric Corporation letter, L. D. White to A. Schwencer, NRC, September 13, 1977.**

4. *NRC Standard Review Plan, Section 5.3.2.**

5. S. E. Yanichko, Westinghouse Report WCAP-7254, May 1969.**

6. T. R. Mager, et al., Westinghouse Nuclear Energy Systems Report FP-RA-1, April 1, 1973.*

7. S. E. Yanichko, et al., Westinghouse Report WCAP-8421, November 1974.**

8. Letter from LeBoeuf, Lamb, Leiby and MacRae to E. G. Case dated March 10, 1975.**

9. NRC letter (A. Schwencer) to Rochester Gas and Electric Corporation dated April 7, 1977.**

*Available at the National Technical Information Service, Springfield, VA 22161.
**Available in NRC PDR for inspection and copying for a fee.
***Available in source file for USNRC Report NUREG-0569.

10. Southwest Research Institute Report, SwRI 17-2376, January 1972.**

11. Southwest Research Institute Reports SwRI 17-3074, July 1972; 17-3377, July 1972; 17-3874, September 1974; and 17-4206, July 1975.**

12. Letter, Rochester Gas and Electric Corporation (L. D. White) to B. C. Rusche dated February 8, 1977.**

13. NRC letter (V. Stello) to Rochester Gas and Electric Corporation dated May 17, 1977.**

14. NRC Report, NUREG-0410 dated January 1, 1978.*

15. NRC letter (A. Schwencer) to Rochester Gas and Electric Corporation dated August 11, 1976.**

16. Rochester Gas and Electric Corporation letter (L. D. White) to A. Schwencer dated July 29, 1977.**

17. Rochester Gas and Electric Corporation letters (L. D. White) to A. Schwencer dated September 3, October 15, and December 8, 1976; and February 24, March 31 and April 26, 1977.**

18. NRC letter (D. Ziemann) to Rochester Gas and Light dated April 18, 1979.**

*Available at the National Technical Information Service, Springfield, VA 22161.
**Available in NRC PDR for inspection and copying for a fee.

APPENDIX H

MILLSTONE NUCLEAR POWER STATION, UNIT 1
REACTOR VESSEL

Design

The nuclear steam supply system of Millstone Nuclear Power Station, Unit 1, was designed by General Electric. It is a model BWR/3 system. The Millstone reactor vessel was designed and fabricated by Combustion Engineering in accordance with ASME Code Section III (1965 Edition) and General Electric's specifications (Refs. 1, 2). In the beltline region the vessel ID is 224 inches and its thickness is 5-11/16 inches. The vessel design pressure is 1250 psig and the normal operating pressure is 1000 psig. The date of commercial operation is March 1971. As of January 1, 1979, Millstone has operated for about 5 EFPY. The maximum estimated end-of-life fluence on the vessel wall ID is 3×10^{18} n/cm^2.

Materials

The reactor vessel is a vertical cylindrical pressure vessel made from SA-302, Grade B, steel modified by the addition of 0.4% to 0.7% nickel. Nickel is added to increase hardenability. With this nickel addition the composition of SA-302, Grade B, steel is identical to that of SA-533, Grade B, Class 1, material (Ref. 3).

The vessel interior is clad with weld deposited E-308 electrode, which results in a cladding of slightly higher nickel and chromium content than Type 304 stainless steel, and identical corrosion resistance to Type 304. The reactor vessel head is flanged to the vessel and sealed with two concentric silver plated, stainless steel, self-energizing O-rings. The area between the two O-rings is vented and monitored to provide an indication of leakage from the inner O-ring seal. The steam outlet lines are from the vessel body, below the reactor vessel flange. The cylindrical part of the vessel is composed of four shell courses; the top, intermediate, lower intermediate and lower courses. Each shell course is made from three plates joined together by vertical seam welds. The courses are then joined by circumferential welds. All welds were made by the submerged arc process using RACO 3-3/16 and nickel 200-1/16 weld wires and Linde 1092 flux. Post-weld heat treatment is carried out at 1150°F ± 25°F with heatup and cooldown rates from and to 600°F of 100°F per hour. The time at temperature for a specific weld depends on its sequence in the vessel's fabrication. Intermediate and final stress relief times are chosen so as to provide 40 hours at 1150°F for the plates in the vessel. A chemical analysis was made on all plate and weld materials. However, the copper content was not reported. The phosphorus content of the weld metals in the beltline region are all 0.021% and the phosphorus content of plate metals varies from 0.008% to 0.011%. The maximum RT_{NDT} value for plate metal was +10°F and for weld metal was +30°F. The upper shelf energy of plate materials in the beltline region varied from 95 to 116 ft-lbs (longitudinal orientation). Charpy tests on weld materials varied from 42 to 82 ft-lbs at a temperature of 10°F. Complete Charpy curves were not performed on weld metals so no upper-shelf energy values are available. These results are considered to be above average for these types of materials.

Based on chemical composition and expected fluence, the limiting vessel material is expected to be weld metal in the longitudinal seams of the lower and lower intermediate shell courses and in the circumferential seam joining these two shell courses.

Material Surveillance Program

Millstone's material surveillance program is based on ASTM E185-66 (Ref. 4). The program consists of three capsules attached to the vessel wall and located opposite the vertical center of the core. Each capsule contains Charpy and tensile specimens made from vessel base, HAZ and weld metals. One capsule contains 12 Charpy specimens of each material type, one contains 9 each, and the third contains 8 each. All specimens are machined from material at the 1/4 T location and are oriented in the longitudinal or strong direction. The base material is from the same heat of plate material used for the lower intermediate shell course. The weld sample simulates the longitudinal weld seams in both the lower and lower intermediate shell courses; i.e., the same heats of weld wire and flux were used to make the welds. Each capsule also contains iron, copper and nickel wire dosimeters that will be used to obtain the neutron fluence. In addition, one capsule basket contains special iron and copper wire that can be removed independently of the surveillance specimens.

Since this program was developed before the initial issuance of Appendix G, 10 CFR Part 50, it does not completely meet all the provisions of this appendix. First, it has only one of its three capsules containing the required number (12) of Charpy specimens. The other two capsules contain nine and eight specimens. This number of specimens will not detract from the program if the test temperatures are prudently selected. Also, it is noted that the Charpy specimens are cut in the strong direction. This discrepancy can be overcome by relating the properties obtained in the strong direction to the weak direction as outlined in NRC Standard Review Plan, Section 5.3.2.

The special dosimeter wires were removed from the reactor vessel during the first refueling outage. From these wires it was determined that the maximum end of life fluence on the vessel wall ID will be 3×10^{18} n/cm^2.

The Millstone Technical Specifications contain no capsule withdrawal schedule. To comply with Appendix H, it is recommended that the first capsule be withdrawn at about 10 EFPY.

Pressure-Temperature Operating Limits

Northeast Nuclear Energy Company (NNECO) in letter dated June 8, 1976, submitted a proposed amendment to the Technical Specifications of Millstone, Unit 1 regarding pressure-temperature operating limits (Ref. 5). In order to complete our review of this proposed amendment, we requested additional information regarding compliance with Regulatory Guide 1.99 and Section IV.A.2.b of Appendix G, 10 CFR Part 50. This section of Appendix G states that the pressure-temperature limits shall provide the same margins of safety in regions of discontinuities as required in the vessel beltline region. Regions of discontinuities, such as nozzles, are not irradiated and hence do not degrade during life. Thus, an analysis of these areas is only required during the first part of life (until such time as radiation causes the beltline region to be limiting by increasing its RT_{NDT} value). Northeast Nuclear Energy Company submitted an analysis of discontinuity areas in the Millstone 1 reactor vessel (Ref. 6). This analysis was reviewed and considered acceptable. Based on this discontinuity analysis revised pressure-temperature limit curves were submitted for our evaluation (Ref. 6). The limiting curve is based on the feedwater nozzle evaluation assuming a

1/4 T flaw and an initial RT_{NDT} of 40°F. A temperature shift of 50°F (resulting from irradiation) is required for the beltline material to become the limiting material. According to Regulatory Guide 1.99, a fluence of 6.75×10^{17} n/cm^2 is required to produce a 50°F shift in the RT_{NDT} of the limiting beltline material. This level of fluence is expected to be reached in the mid-1980s when the projected integrated core exposure is 9×10^6 MWD_{th}.

From the review by the technical staff it was concluded that the proposed pressure-temperature operating limit curves are acceptable and are in conformance with the rules of Appendix G, 10 CFR Part 50 (Ref. 7). For operation through 9×10^6 MWD_{th}, the feedwater nozzle material will be limiting. By the time these operating limits are required to be revised, a material surveillance capsule will have been tested. The results of tests on this capsule will provide a bases for radiation damage needed to calculate the next set of operating limits. The proposed operating limits were incorporated into the Technical Specifications as Amendment No. 62 dated June 1, 1979.

Inservice Inspection Program

A preoperational examination was conducted over a period of 1½ years by Southwest Research Institute (Ref. 8). There were three different examination periods: April 1969, February 1970, and August 1970. Between the February and August inspections furnace sensitized stainless steel safe ends were either replaced or cladded. Following these modifications, the affected areas were reexamined. Both mechanized and manual ultrasonic techniques were used. Generally, welds were inspected by the technique that was expected to be used for inservice inspections. However, some welds were inspected by both techniques. The only significant indications found were caused by geometric conditions such as those reported in the head nozzles, steam outlet nozzles, and core spray nozzle. The vessel cladding was also surface inspected with dye penetrant and no indications observed. These examinations are used as a baseline for future inservice examinations.

Through 1978, five inservice examinations have been conducted; 1972, 1974, 1975, 1976 and 1978 (Ref. 9). The first four inspections were conducted in accordance with the 1971 Edition of Section XI. The 1978 examination was conducted in accordance with the 1974 Edition of Section XI and Addenda through Summer 1975. To date, some welds of each inspection category except Category B-A welds have been examined. The only unacceptable indications reported were in the 1976 inspection: flaws in several closure nuts and a surface flaw in one nozzle to safe end weld.* The subject area on the safe end weld was polished and reexamined. The retest showed no reportable indication.

On September 1, 1972 sea water entered the reactor system as a result of pin holes that developed in the condenser tubing. As a result of this sea water intrusion, a program was initiated to determine the effects of this intrusion on various reactor components (Ref. 10). The program consisted of stress corrosion tests on applicable materials and augmented inservice examinations of affected components. The 1972 inservice inspection was expanded to include most of the reactor vessel welds, and was used as a baseline for future examinations to be conducted as part of the program. Inservice examinations were required after six weeks and five months of operation and at the next refueling outage. These examinations showed no evidence of flaw growth as a result of this incident.

*Inspection results on CRD RL and feedwater nozzles are discussed in the "Generic Safety Item" section.

To comply with the requirements of 10 CFR 50.55a(g), Northeast Nuclear Energy Company submitted an updated inservice inspection program for the third Millstone inspection period of the first 10-year interval (Ref. 11). Since the vessel was designed prior to the initial issuance of Section XI, relief from examination of several welds was requested. The most important category where relief was approved is the Category B-A welds in the beltline region (Ref. 12). The basis for this relief is that the reactor vessel is insulated with permanent reflective insulation and surrounded by the concrete biological shield. The annular space between the inside diameter of the biological shield and the outside diameter of the insulation is a nominal 6½ inches. Thus, access for removal of the insulation panels is extremely limited and this inaccessibility precludes direct examination of these welds from the outside surface. The interior surface of the reactor vessel is stainless steel clad and the vessel's internals, shroud, and jet pumps would make an internal volumetric examination of most of these welds impractical. Also, the staff feels that other methods of volumetric examination with the existing limitations which will produce meaningful results have not been fully developed at this time. However, acoustic emission, which is considered to be in the developmental stage, is being studied by the licensee as an examination technique for these welds. The licensee has committed to adopt acoustic emission and include it in the inservice inspection program for surveillance of these welds when a system is demonstrated to be practical and endorsed by the NRC.

Other welds where relief was granted are the Category B-H vessel support welds and Category B-O CRD housings. In the case of the vessel supports, the staff has recommended that a surface examination be substituted for the required volumetric examination. This program was incorporated into the Technical Specifications by Amendment No. 64 dated September 19, 1979.

The 1978 inservice inspection was conducted in accordance with the revised program above. This was the first inspection of the third inspection period of the first inspection interval. This first 10-year inspection interval is scheduled to be completed in December 1980.

Generic Safety Items

The generic safety items applicable to Millstone 1 are reactor vessel low upper shelf energy, sensitized stainless steel safe ends, CRD RL nozzle cracking and feedwater nozzle cracking.

Since no material surveillance capsules have been removed from the vessel and tested to date, the low upper shelf energy item can not be completely resolved at this time. However, because the fluences on the vessel in the beltline region are estimated to be low, it is unlikely that this vessel will have a low upper shelf problem during its service life. This item will be reevaluated when the results of tests on the first surveillance capsule removed from the vessel are reviewed.

As originally constructed, the Millstone 1 primary coolant system and ECCS had 22 furnace-sensitized stainless steel safe ends (Ref. 13). Prior to initial operation, the safe ends on the recirculation outlet nozzles were clad with 308L stainless steel to a minimum thickness of 0.06 inches. All other sensitized safe ends were replaced with solution treated 304 stainless steel. We conclude that the above corrective action is acceptable and has corrected the Millstone 1 sensitized stainless steel safe end problem (Ref. 14) This conclusion is supported by the results of subsequent inservice examinations on these safe ends which have revealed no flaws.

Millstone 1 has had a history of feedwater nozzle cracking and sparger failures (Ref. 15). From 1972 to 1975, four design modifications were made on these spargers. Following each of these modifications further cracking was detected. Flaws as deep as ½ inch have been reported. Then, in the winter 1976 refueling outage, all feedwater spargers were replaced with spargers having thermal sleeves with an interference fit (Ref. 16). This is the same design as used on Dresden 2/3 and Quad Cities 1/2. Prior to the new installation all nozzles were inspected externally by ultrasonic methods and internally by dye penetrant and all indications were removed by grinding. This design was Design Number 5 (Ref. 17). To further reduce thermal stresses in the nozzle area and thus reduce the probability of cracking, NNECO plans to remove the nozzle cladding in 1980. An inspection was conducted on the feedwater nozzles in 1978 and no indication of further cracking was found. A program to monitor the feedwater nozzles was submitted to NRC (Ref. 18). We believe that this latest sparger design should eliminate cracking in the feedwater nozzles. The proposed inspection program will adequately monitor these nozzles for any signs of further cracking or deterioration.

Cracks have also been detected in the Millstone 1 CRD RL nozzle area. The licensee has taken steps to prevent further cracking. Corrective measures are based on the recommendations contained in General Electric SIL No. 200, Supplement 2, and NUREG-0312 (Refs. 19,20). In 1978 NNECO rerouted the CRD RL to a feedwater line. This rerouted line will be operated in a continuous flow mode (Ref. 17). Also, the existing drywell penetration line previously associated with the return line has been capped to eliminate cold water flow through the nozzle. The vessel nozzle was ultrasonically examined in 1978 from the reactor vessel exterior. During either the 1979 or 1980 refueling outage, NNECO plans to remove the CRD RL thermal sleeve and the stainless steel return line piping and safe end. The nozzle will then be capped. Finally, the nozzle bore, blend radius and vessel wall adjacent to the nozzle will be examined with dye penetrant from the vessel interior and any indications found will be removed by grinding. The NRC staff has reviewed this program and concludes that it is acceptable and will provide assurance that further cracking will not occur.

Conclusions and Recommendations

The Millstone 1 reactor vessel was designed to the 1965 Edition of ASME Code Section XI. A complete preservice nondestructive examination was performed on components of the reactor vessel and no significant indications were reported. From our review of the design criteria and the results of the preservice examination, it is concluded that the initial integrity of the reactor vessel is acceptable. Inservice inspections have been performed on the vessel in accordance with ASME Code Section XI rules since 1972. Except for flaws found in one feedwater nozzle safe end, and in the CRD RL and feedwater nozzles, no serious flaws have been detected. The reactor vessel is operating with pressure-temperature operating limits that are in accordance with Appendix G, 10 CFR Part 50. The staff will continue to review and update these limits to account for further radiation damage on vessel beltline materials. The amount of radiation damage will be predicted from Regulatory Guide 1.99 until about 10 EFPY at which time the first Millstone 1 surveillance capsule will be removed from the vessel. After 10 EFPY the results of tests on this capsule will be used to obtain radiation damage. The Millstone 1 material surveillance program has been reviewed and is considered acceptable.

Based on the unirradiated mechanical properties of the vessel materials, it is concluded that these materials will have acceptable fracture toughness properties throughout service life. However, this topic will be reviewed again when the staff reviews the results of the surveillance capsule to be pulled at 10 EFPY. This is a normal part of the staff review of surveillance results. We

conclude that the combination of inservice inspections, conservative pressure-temperature operating limits and the use of materials having acceptable fracture toughness provides assurance that the vessel integrity will be maintained at an acceptable level throughout service life. The generic safety items applicable to Millstone 1 (low upper shelf toughness, sensitized stainless steel safe ends, and CRD RL and feedwater nozzle cracking) have been successfully resolved for the purposes of this report and will not adversely affect the integrity of the reactor vessel. NRC Technical Activity A-10 will complete the resolution of the feedwater and CRD RL nozzle issues.

The following recommendations are made:

1. The Technical Specifications do not contain a material surveillance capsule withdrawal schedule. It is recommended that a withdrawal schedule be put in the Millstone 1 specifications. The first capsule should be withdrawn at about 10 EFPY, a second at about 18 to 24 EFPY and the third should be a standby capsule.

References (Appendix H)

1. Millstone, Unit 1, FSAR.**

2. Amendment 13, Appendix E, to Millstone FSAR.**

3. NNECO letter (W. Fee) to Director, NRR, dated July 31, 1978.**

4. J. Higgins and F. Brandt, General Electric Report, NEDO 10115, July 1969.**

5. Letter, NNECO (W. Fee) to Director, NRR, dated June 8, 1976.**

6. Letter, NNECO (D. Switzer) to Director, NRR, dated July 1, 1977.**

7. NRC internal memorandum, L. Shao to K Goller dated August 25, 1977.***

8. Southwest Research Institute Final Report SwRI Project 17-1975, February 1972.**

9. General Electric (Installation and Services Operations) Reports dated September 1, 1972, April 22, 1975, December 22, 1975, January 27, 1977, and August 15, 1978.***

10. Millstone Point Company Special Report, "Chloride Intrusion Incident," including Appendices A though I, December 11, 1972.**

11. Letters, NNECO (D. Switzer) to Director, NRR, dated February 28, 1977 and May 27, 1977.

12. NRC internal memorandum, L. Shao to K. Goller dated January 23, 1978.***

**Available in NRC PDR for inspection and copying for a fee.
***Available in source file for USNRC Report NUREG-0569.

13. Amendments 9, 25 and 28 to the Millstone 1 FSAR.**

14. USAEC Addendum to the Safety Evaluation of Millstone 1 dated October 7, 1970.**

15. Millstone Interim Report, Addenda 4, "Feedwater Sparger Failure of Design 3 and Correlation of Findings with Design 4," October 2, 1974.**

16. Letter, NNECO (D. Switzer) to Director, NRR, dated October 1, 1976.**

17. Letters, NNECO (D. Switzer) to Director, NRR, dated November 18, 1977 and March 31, 1978.

18. Letter, NRC (G. Lear) to NNECO dated March 10, 1978.**

19. U.S. Nuclear Regulatory Commission Report, NUREG-0312 dated July 1977.*

20. General Electric SIL No. 200, Supplement No. 2, dated November 18, 1977.**

*Available at the National Technical Information Service, Springfield, VA 22161.
**Available in NRC PDR for inspection and copying for a fee.
***Available in source file for USNRC Report NUREG-0569.

APPENDIX I

DRESDEN NUCLEAR POWER STATION, UNIT 2
REACTOR VESSEL

Design

The nuclear steam supply system for Dresden Nuclear Power Station, Unit 2, a two-loop recirculation system, was designed by General Electric. The reactor vessel was designed and fabricated by Babcock and Wilcox in accordance with General Electric specifications and the rules of ASME Code Section III, 1963 Edition and Addenda through Summer of 1964 (Ref. 1). The General Electric requirements are contained in specifications 21A1109AB, Rev. 12 dated March 11, 1969, and 21A1109, Rev. 2, dated September 24, 1968 (Ref. 2). This is a model BWR/3 system similar to that of Quad Cities 1 and 2.

In the vessel beltline region the reactor vessel ID is 251 inches and its thickness is 6-1/8 inches. The inner surface of the vessel is clad to a minimum thickness of 1/8 inch with ASTM A371, Type ER 308L stainless steel weld overlay. The design and operating pressures of the vessel are 1250 psia and 1000 psia, respectively. The date of commercial operation is June 9, 1972. As of January 1, 1979, Dresden 2 has operated for about 5 EFPY. At end of life the maximum fluence on the vessel wall ID is estimated to be 1×10^{18} n/cm^2.

Materials

The reactor vessel consists of a cylidrical section and a top and bottom head (Ref. 2). The cylindrical section is composed of four shell courses. The shell courses are joined together by submerged metal arc welds. Each of the four shell courses is formed from SA-302, Grade B, steel plate material modified by the addition of 0.3% to 0.7% nickel. This nickel is added to increase the hardenability of the material. The nickel additive makes this material identical to SA 533, Grade B, Class 1, material. The plate material was purchased from Lukens steel. The two intermediate shell courses consist of four plates and the top and bottom courses consist of three plates. These plates are welded together with electroslag welds. This is the first nuclear reactor vessel fabricated with electroslag welds. The AEC, therefore, required a great deal of information on the welding technique and the quality assurance measures used during fabrication. The licensee's responses are presented in Reference 3. These responses were reviewed by the AEC staff as part of their license review and were again reviewed as part of this present SEP review. It is concluded from these reviews that the techniques used to fabricate these welds are satisfactory.

In a letter from NRC (D. K. Davis) to Commonwealth Edison dated August 26, 1977, NRC requested information on the reactor vessel materials in the vessel beltline region and on the materials in the material surveillance program. The staff has not received an adequate response to this request. Until this response is submitted, we cannot complete the review of the vessel materials.

Material Surveillance Program

The Dresden 2 material surveillance program consists of six wall capsules and three accelerated capsules (Refs. 3,4). The program includes tensile and Charpy specimens made from base, and both electroslag and submerged arc weld and HAZ material. All specimens are machined from material at the 1/4 T location. Specimens from the plate material are oriented in the longitudinal direction (strong direction). All the capsules contain specimens from the electroslag weld. All but one wall capsule contain specimens from the base metal. All but two wall capsules contain specimens from the submerged arc weld material. The weld material used in this program simulates that used in the vessel. However, until an acceptable response to NRC letter dated August 26, 1977, is received (see Materials Section above), we cannot determine if the surveillance weld metal is identical to the beltline weld metal, i.e., was made from the same weld wire and flux. The base material used in the surveillance program was obtained from the material left over from one of the vessel beltline plates. Capsules also contain iron, copper and aluminum-cobalt dosimeter wires.

Since the program was designed prior to the initial issuance of Appendix H, 10 CFR Part 50, it does not meet all the requirements of this appendix. However, the program does have more than the required number of capsules. Thus, although some of these capsules do not contain all the vessel beltline materials, there still is a sufficient number of capsules containing all beltline materials to meet Appendix H requirements. The main areas where the program does not conform to Appendix H is the number of specimens and the orientation of the Charpy base metal specimens. Capsules contain either eight or nine Charpy specimens from each type of material. The Appendix H requirement is 12 Charpy specimens from each material. Capsules also contain only two tensile specimens from each material instead of the three required by Appendix H. The Charpy base metal specimens are oriented in the strong direction instead of the weak direction required by Appendix H. However, these limitations are not considered serious enough to limit the effectiveness of the program to predict radiation damage on the vessel beltline materials. Therefore, the program is considered to be acceptable.*

The unirradiated properties of the surveillance materials are reported in Reference 5. A chemical analysis was made on the surveillance materials. The cooper content of the plate, submerged metal arc and electroslag materials is 0.19%, 0.18% and 0.19%, respectively. To date four capsules have been removed from the vessel. Tests have been completed on three of these capsule specimens. The specimens from the fourth capsule are currently being tested at Battelle Columbus Laboratories. This fourth capsule is an accelerated capsule.

Two of the three capsules that have been tested are accelerated capsules. The third capsule, a wall capsule, received a fluence of only 1.3 x 10^{16} n/cm^2 (Ref. 6). At this low fluence level no measurable changes in the mechanical properties were reported. From the test results of other surveillance programs and the research programs, this result is expected. In the first accelerated surveillance capsule pulled, base, submerged arc weld and electroslag weld specimens received average fluences of 9.5, 6.4 and 9.5 x 10^{18} n/cm^2, respectively (Ref. 6). These test results are significant because these fluences are slightly above the fluence expected on the vessel materials at end of life. The upper shelf energy of the base metal dropped from 153 to 117 ft-lbs and its

*This acceptability is based on the assumption that the surveillance weld materials were made from the same type of weld wire and flux as those used to fabricate the reactor vessel. It is exemplary that this surveillance program contains materials from both types of welds used in vessel fabrication. However, for this data to be fully utilized, the surveillance material must be correlated to the vessel weld materials.

RT_{NDT} at 30 ft-lbs increased by 40°F. The upper shelf energy of submerged arc weld material dropped from 71 to 51 ft-lbs and its RT_{NDT} increased by 65°F. For electroslag metal the upper shelf energy decreased from 100 to 80 ft-lbs and RT_{NDT} was increased by 100°F. Specimens in the second accelerated capsule tested were subjected to fluences in excess of 1×10^{19} n/cm^2 (Ref. 7). Since this fluence is well above that anticipated on the vessel wall at the end of life, the results of these tests will not affect the future operation of Dresden 2. One noteworthy result from these tests is that the upper shelf energy of submerged arc material dropped to only 70 ft-lbs.

The next capsule is scheduled to be removed from the vessel at about 8 EFPY.

During this review, it came to our attention that Commonwealth Edison had submitted a request for a change to Dresden 2 Technical Specifications regarding the surveillance capsule removal schedule in letter dated May 16, 1977. No action was taken by the NRC staff on this request until November 1978, at which time this item was called to the attention of the Dresden 2 Project Manager and a request to review this proposed Amendment was issued. The proposed withdrawal schedule is based on the three capsule schedule, all of which are wall capsules. Accelerated capsules are considered to be additional capsules. This revised withdrawal program is considered to be in accordance with the requirements of Appendix H and is, therefore, acceptable. It was incorporated into the Technical Specifications as part of Amendment 44 dated August 13, 1979.

Pressure-Temperature Operating Limits

As part of the review of Dresden 2 application for a Full Term Operating License, the pressure-temperature operating limits were reviewed in 1974 (Ref. 9). It was found that the limits in the Technical specifications were not in conformance with Appendix G, 10 CFR Part 50. Commonwealth Edison, therefore, submitted a proposed Ammendment to the Technical Specifications in letter dated September 10, 1974 (Ref. 10). These proposed limits were reviewed and considered acceptable by the NRC technical staff. However, these limits were never incorporated into the Technical Specifications (probably due to the fact that the FTL review was terminated and no Full Term Operating License was granted). As part of the present SEP review, the operating limits were again reviewed. Since a review of a proposed amendment to revise the operating limits of Quad Cities 1 and 2 had just been completed, it was suggested that these revised limits should also be used for Dresden 2. The reactor vessels of these three plants are almost identical. The proposed operating limits for Quad Cities 1 and 2 contained limit curves based on an analysis for regions remote from the beltline, including areas of discontinuities, and for the beltline region (Refs. 11,12). At the beginning of life regions remote from the beltline are limiting. These areas do not receive any significant amount of radiation. On the other hand, the beltline region materials do receive a significant amount of radiation. Thus, after some period of operation the beltline materials will become the controlling materials. The period of operational time before the beltline materials will become limiting was estimated from the results of the data on the Dresden 2 material surveillance program. We concluded that this would be in 6 EPFY. This estimate was based on an end of life fluence value at the 1/4 T location on the vessel wall of 9×10^{17} n/cm^2. This is the "worst case" fluence value obtained from Reference 13. These operating limits were incorporated into the Technical Specifications as License Amendment 39 (Ref. 14). Since these limits are acceptable for operation through 6 EFPY, they will need to be revised in the next several years for Dresden 2. The next operating limit curves for Dresden 2 should be based on the irradiated properties of the vessel's beltline materials. From the surveillance data reviewed to date, it appears that the limiting material will be one of the weld metals in the vessel beltline region.

The present operating pressure-temperature limits in the Technical Specifications are acceptable for operation through 6 EFPY and are in conformance with Appendix G, 10 CFR Part 50 for this time period.

Inservice Inspection Program

A preoperational inspection was performed on the reactor vessel in July 1969 by General Electric (Ref. 15). The examinations were conducted in accordance with the 1968 draft to ASME Code Section XI. Most of the accessible welds in the vessel were inspected. The only notable indications found were 11 indications in the head to flange weld. These indications were attributed to weld porosity. The results of these examinations will be used as a baseline for reference to future examination results.

General Electric also conducted inservice examinations on the reactor vessel in the spring of 1971 and in the winter of 1972 (Ref. 16). In the 1971 inspection, indications were noted in the core spray nozzle safe end. In the 1972 inspection, indications were detected in several nozzle to safe end welds. All of the above flaws were removed by grinding. Examinations were also conducted in November 1974 to July 1975, spring 1975 and fall 1977 (Ref. 17). These three examinations were conducted in accordance with the ASME Code Section XI, 1971 Edition and Addenda through Summer 1971. In the 1974-75 inspection, indications were found in the feedwater nozzle cladding. These were removed by grinding. In the 1976 examination, indications were also noted in the feedwater nozzle cladding. Although the above examinations were conducted in accordance with Section XI rules, the examination of vessel safe ends were conducted at a greater frequency than specified in the Code. These augmented inspection requirements were required by the AEC staff because of concerns over the use of sensitized stainless steel in the safe ends (Ref. 18). To date, some welds of each inspection category listed in Section XI have been examined with the exception of Category B-A welds in the vessel beltline region. The first 10-year inspection interval is scheduled to terminate in February 1981.

In letter dated July 31, 1978, Commonwealth Edison submitted a revised inservice inspection program as required by 10 CFR 50.55a(g) (Ref. 19). This program conforms to the 1974 Edition of Section XI and Addenda through Summer 1975. The licensee requested relief from examining all Category B-A welds and some Category B-B welds because of lack of accessibility. Inspection of these welds from outside the vessel is precluded due to the vessel insulation and the close proximity of the biological shield wall. The mirror-type insulation on the vessel wall is made up of interlocking panels making removal difficult. The annular dimension between the shield wall and the insulation is only four inches which is not sufficient to allow the use of remote inspection equipment. Inspection of these welds from inside the vessel is hampered by the vessel internal design. The core shroud, jet pumps, and other brackets are not designed to be removed. To clarify some of the relief requests and to make certain changes required by the staff to the program, a revised program was submitted to NRC on June 26, 1979. In this revised program, the licensee requested an additional relief from inspecting the reactor vessel cladding. The staff has not completed their review of this program. However, based on previously reviewed programs, it is expected that the relief requests for examination of the Category B-A and B-B welds will be granted. The staff review is expected to be completed by fall 1979.

Generic Safety Items

The generic safety items applicable to Dresden 2 are low upper shelf toughness, sensitized stainless steel safe ends, feedwater nozzle cracking and control rod drive return line nozzle cracking.

Materials in the Dresden 2 accelerated material surveillance capsules, exposed to irradiation higher than that expected on the vessel wall at end of life, have been tested. Results show that these materials will have an upper shelf Charpy energy higher than 50 ft-lbs at the fluence level expected at the end of the vessel life. Therefore, it is concluded that the Dresden 2 vessel materials will not have a low upper shelf energy problem throughout the vessel's service life.

There were originally 27 furnace-sensitized stainless steel safe ends on the Dresden 2 reactor vessel. All of these safe ends were made of either Type 304 or 316 stainless steel. Four of these safe ends have been replaced because stress corrosion cracks were detected in them, and one has been replaced because of a fatigue crack in an adjacent pipe. Thus, there are currently 22 sensitized safe ends on the reactor vessel.

Because of cracking problems on safe ends of the Nine Mile Point and Oyster Creek reactor vessels, both the AEC staff and the ACRS devoted a considerable amount of time in reviewing the Dresden 2 safe end situation during the operating license review. The licensee stated that at the time the Dresden system was erected and cleaned, plant management was aware of the Oyster Creek sensitized stainless steel problem. Therefore, procedures were specifically developed for the Dresden 2 erection and cleaning to prevent the contamination of sensitized surfaces with chlorides and fluorides and to provide for thorough cleaning of these surfaces (Ref. 21). Thus, the Dresden 2 system was installed relatively clean. The licensee also reviewed the stresses on the sensitized safe ends and performed a fatigue analysis based on ASME Code Section III using the fatigue strength of sensitized material (Ref. 21). Stresses were found to be very low. The usage factors calculated in the fatigue analysis were also low; safety margins were at least five. The staff concluded that the Dresden 2 safe ends did not need to be replaced provided that augmented inservice examinations were performed on these components (Refs. 22, 23, 24). The Dresden 2 safe ends have been inspected in the summer and in the winter of 1970, spring 1971, spring 1972, 1974/1975, spring 1976 and fall 1977. Except for flaws noted in the safe ends that have been replaced, the safe ends appear to be in good condition. The only indications detected, other than those in the replaced safe ends, have been small and have been removed by grinding. From this present review, it is recommended that an augmented inspection program be continued. It is also concluded that these safe ends have acceptable integrity and are acceptable for further operation. How long they will be acceptable will depend on the results of future inservice inspections.

As originally designed, Dresden 2 feedwater nozzles were fitted with loose fit, Tee box spargers. During the November 1974 outage, these spargers were replaced with forged Tee spargers with an interference fit thermal sleeve (Ref. 25). These redesigned spargers are similar to those installed on Dresden 3 and Quad Cities 1 and 2. Following this modification, the blend radii of all four nozzles were inspected using dye penetrant (Ref. 26). Over 400 indications were found. These indications were removed by grinding. During the spring 1976 outage, the accessible parts of the nozzles were inspected by ultrasonic and dye penetrant techniques (Ref. 26). Several small indications were noted. In the fall 1977, a dye penetrant examination was performed on the accessible areas of three of the feedwater nozzles and a complete examination was performed on the fourth nozzle with the sparger removed (Ref. 26). The only flaws detected were on the nozzle with the

sparger removed where nine indications, the largest being less than 1/16-inch deep, were detected. It is probable that these cracks existed in the material prior to the installation of the new spargers. All indications were removed by grinding. To provide a "final fix" solution to the feedwater nozzle cracking problem, Commonwealth Edison plans to install the new General Electric double seal/triple thermal sleeve sparger and remove the feedwater nozzle cladding. This work is scheduled to be performed in the fall 1980 outage (Ref. 27). Augmented inservice inspections will continue to be required. The staff considers that the above actions by the licensee provide an acceptable resolution to the Dresden 2 feedwater nozzle cracking problem. The staff will continue to carefully review the results of inservice examinations to make sure there is no further cracking.

The control rod drive return line nozzle is fitted with a thermal sleeve welded in place and having a flange that covers the blend radius. During recent inservice examinations, cracks were found in the sleeve and nozzle safe end (Ref. 28). These cracks were ground out. This line was then valved out, terminating the flow of cold water into the reactor vessel during normal operation. The elimination of this cold water flow will terminate the thermal stresses in this area. Since thermal stresses are believed to be the mechanism responsible for the initial stages of crack growth, the NRC staff concludes that this solution is acceptable. Again the staff will review the results of future inservice inspections to make sure that further cracking has not occurred.

Conclusions and Recommendations

The Dresden 2 reactor vessel was designed to ASME Code Section III. The requirements of this code were supplemented by the requirements of General Electric's purchase specifications that required quality control measures above those required by Section III. To further verify the initial integrity of the reactor vessel, a complete preservice examination was performed on the reactor vessel. From our review of the design criteria, material tests and preservice examination results, it is concluded that the initial integrity of the Dresden 2 reactor vessel is acceptable. Inservice inspections have been performed on components of the vessel since 1971. Since 1974 these inservice examinations have been performed in accordance with the rules of ASME Code Section XI. To date the only major unacceptable flaws detected have been on safe ends. These safe ends have been replaced. The reactor vessel is currently operating with pressure-temperature operating limits that are in accordance with Appendix G, 10 CFR Part 50. The staff will continue to review and update these operating limits to account for additional radiation damage on vessel materials. The amount of radiation damage will be determined from the results of tests on Dresden 2 material surveillance specimens. The Dresden 2 material surveillance program has been reviewed and is considered to be acceptable. To date, three material surveillance capsules have been removed from the reactor vessel and tested. The results of these tests show that the vessel materials will have adequate toughness throughout their service life. The combination of these inservice inspections, conservative operating limits and the use of materials having acceptable fracture toughness provides assurance that the vessel integrity will be maintained at acceptable levels throughout service life. The generic safety items applicable to Dredsen 2 (low upper shelf energy, sensitized stainless steel safe ends, feedwater nozzle cracking and control rod drive return line nozzle cracking) have been successfully resolved for the purposes of this report and will not adversely affect the vessel integrity. NRC Technical Activity A-10 will complete the resolution of the feedwater and CRD RL nozzle issues.

The following recommendations are made:

1. The staff has not received an acceptable response to our request for information on reactor vessel materials and surveillance materials. The licensee should be requested to supply this information in order that we can verify the acceptability of these materials.

2. The next material surveillance capsule is scheduled to be removed at 8 EFPY. Since test results on two previously removed capsules have shown a large difference in the upper shelf energy obtained from Charpy test on submerged arc weld metal, special attention should be paid to obtaining the upper shelf energy of this material on the capsule removed at 8 EFPY.

3. Dresden 2 has 22 sensitized stainless steel safe ends on the reactor vessel. It is recommended that an augmented inservice inspection program be continued on these safe ends. This item should be considered by the staff as they review the proposed Dresden 2 inservice inspection program submitted by the licensee on July 31, 1978.

References (Appendix I)

1. Dresden 2 and 3, Safety Analysis Report, Section IV.*

2. Dresden 2 and 3, Safety Analysis Report, Apendix D.*

3. Dresden 2 and 3, Safety Analysis Report, Appendix F.*

4. General Electric Report NEDO 10115, July 1979.*

5. G. F. Rieger and G. Henderson, General Electric Report NEDC 12575, April 1975.*

6. G. Rieger and G. Henderson, General Electric Report NEDC 12585, May 1975.*

7. Battelle Columbus Report BOL-585-3, September 15, 1977.*

8. Letter, Commonwealth Edison (R. Bolger) to Director, NRR, dated May 16, 1977.*

9. AEC letter (D. Skovholt) to R. Maccary dated February 22, 1974 (internal communications).**

10. Commonwealth Edison letter (B. Lee) to Directorate of Licensing, AEC, dated September 10, 1974.*

11. Commonwealth Edison letter (R. Bolger) to Director, NRR, dated May 17, 1976.*

12. Commonwealth Edison letter (M. Turbak) to NRC, CRB-3, dated March 13, 1978.*

13. General Electric Report, J. N. Kass, et al., NEDO-21708, October 1977.*

*Available in NRC PDR for inspection and copying for a fee.
**Available in source file for USNRC Report NUREG-0569.

14. NRC letter (T. Ippolito) to Commonwealth Edison dated November 13, 1978.*

15. General Electric Report NEDO 10107, August 1970.*

16. General Electric Reports NEDC 10470 dated December 1971 and NEDO 13288 dated 1973.*

17. Commonwealth Edison letters (B. Stephenson) to Division of Boiler Inspection, State of Illinois, dated September 29, 1975; August 9, 1976 and February 17, 1978.**

18. AEC letter (P. Morris) to Commonwealth Edison dated June 7, 1971.*

19. Commonwealth Edison letter (M. Turbak) to NRC, ORB-3, dated July 31, 1978.*

20. Commonwealth Edison responses to ACRS concerns dated May 22, 1970 and June 5, 1970.**

21. Commonwealth Edison letter (W. Stiede) to AEC (P. Morris) dated December 13, 1971.*

22. AEC letter (D. Skovholt) to Commonwealth Edison dated January 8, 1973.*

23. ACRS letter (J. Hendrie) to AEC (G. Seaborg) dated July 17, 1970.*

24. AEC letter (P. Morris) to Commonwealth Edison dated July 24, 1970.*

25. Commonwealth Edison letter (M. Turbak) to NRC (ORB-3) dated October 27, 1978.*

26. Commonwealth Edison letters (M. Turbak) to NRC (ORB-2) dated November 28, 1977 and May 12, 1976.*

27. Commonwealth Edison letter (M. Turbak) to NRC (ORB-2) dated December 5, 1978.*

28. Commonwealth Edison letter (M. Turbak) to NRC (ORB-2) dated April 25, 1978.*

*Available in NRC PDR for inspection and copying for a fee.
**Available in source file for USNRC Report NUREG-0569.

APPENDIX J

PALISADES NUCLEAR GENERATING STATION REACTOR VESSEL

Design

The nuclear steam supply system for Palisades Nuclear Generating Station, comprised of two heat transfer loops, was designed by Combustion Engineering. The reactor vessel was designed and fabricated by Combustion Engineering in accordance with their specifications and to the rules of ASME Code Section III and Addenda through Winter 1965 (Ref. 1).

During design and fabrication of the reactor vessel, a number of examinations over and above the requirements of the ASME Code, Section III, were performed by the vendor (Ref. 1). For example, dye penetrant examinations were conducted on the vessel cladding, CRD head nozzle connection welds and instrumentation nozzles. Ultrasonic examinations were performed on vessel cladding and vessel support buildup welds. Other quality control measures utilized by the vendor that were not required by Section III include the preparation of detailed purchase specifications which included cooling rates for test samples; requirement for vacuum degassing for all ferritic plates and forgings; specification of fabrication instructions for plates and forgings to provide control of material prior to receipt and during fabrication; use of written instructions and manufacturing procedures which enabled continual review based on past and current manufacturing experiences; performance of chemical analysis of welding electrodes, welding wire, and materials for automatic welding, thereby providing continuous control over welding materials; and the determination of NDT temperature through use of drop weight testing methods as well as Charpy impact tests.

In the beltline region the reactor vessel ID is 172 inches and its thickness is 8½-inches. The inner surface of the vessel is clad to a minimum thickness of 3/16-inch with Type 304 stainless steel. The design and operating pressures are 2500 psia, and 2100 psia (initial) and 2250 psia (stretch), respectively. The date of commercial operation is December 21, 1971. As of January 1, 1979, Palisades has operated for about 3 EFPY.

Materials

The cylindrical section of the vessel consists of three shell courses: the lower, the intermediate and the upper shells. The upper shell course contains the primary vessel nozzles. Each shell is formed from three plates welded together by vertical seams. All plates are made of SA-302B steel modified by the addition of about ½% nickel (Ref. 2). This nickel additive makes the SA-302B steel identical to SA-533, Grade B, Class 1, material. The three cylinders were joined together with two circumferential welds. All welds were made by the submerged arc process using a dual wire feed consisting of a Mil B-4 wire (RACO 3 3/16) and a 1/16-inch nickel-200 additive, and Linde 1092 flux. The root welds were chipped back and manually welded with 3/16-inch E-8018 wire. Post-weld heat treatment was at 1150°F ± 25°F. The time at temperature for a specific weld depended on its sequence in the vessel's fabrication. Intermediate and final stress relief times were chosen to

provide 40 hours at 1150°F for the plates in the vessel. Thus, the longitudinal seams in the intermediate and lower shell assemblies had 40-hour stress relief treatments while the closing girth seam saw only the final vessel assembly stress relief treatment of 18 hours at 1150°F.

Based on the fluence levels to be received and material chemistry, the limiting vessel materials are the intermediate shell plates and welds. The estimated maximum fluence on these materials at the ID surface at end-of-life is predicted to be 3.8×10^{19} n/cm^2.* The copper content of these materials is 0.25% for plate metal and 0.26% for weld metal. The upper-shelf energy of the plates ranges from 86 to 91 ft-lbs in the weak direction. No upper-shelf energy values were established for the weld materials other than that obtained on the surveillance weld which was 120 ft-lbs. These shelf values for both plate and weld metal are well above average values and are considered acceptable. The maximum value of RT_{NDT} of plate material in the beltline region is 10°F. No dropweight tests were conducted on weld material so RT_{NDT} cannot be calculated. However, based on Charpy data, the RT_{NDT} of weld metal is estimated to be below 0°F.

Material Surveillance Program

The material surveillance program for Palisades is based on ASTM E-185-66 (Ref. 3). The program consists of 10 surveillance capsules placed in three locations with respect to flux levels. One series of capsules is placed adjacent to and on the outside of the core support barrel to obtain an accelerated exposure. These specimens will receive the design lifetime neutron exposure in a relatively short time and will provide data for predicting the NDT temperature shift for the pressure vessel material over the design life of the vessel. A second series of specimens is contained in six capsules and located outside of the core support barrel near the pressure vessel wall. These specimens will receive, at any given time, a slightly higher neutron dose than the pressure vessel. The NDT temperature shifts resulting from the irradiation of these specimens will closely approximate the NDT temperature shift of the vessel materials and will serve as a check on the data obtained from the accelerated exposure specimens. The third series of specimens is contained in two capsules and located in a low flux region above the core. These specimens will be exposed to all reactor temperature cycles but will receive a very low neutron dose. Changes in the mechanical and impact properties of the vessel materials due to thermal exposure only can, therefore, be monitored on the basis of changes in properties of these specimens.

Each capsule contains 3 tensile specimens and 12 Charpy specimens from base, HAZ and weld materials. The base metal Charpy specimens were oriented in the longitudinal direction. Six of the capsules also contain 12 Charpy specimens of base material that were oriented in the transverse direction. All specimens were made from material at the 1/4 T location. The base material for the reactor vessel materials surveillance program is intermediate shell plate No. D-3803-3, piece No. 112-04B (Ref. 2). The weld material was fabricated by welding two intermediate shell plates together using the detailed weld procedure used for the longitudinal weld seams. The same type of filler wire and flux was used in the fabrication of the surveillance material but not the same heat of wire or batch of flux.

Correlation monitor specimens are also included in the surveillance capsules (Charpy V-notch specimens machined from ASTM standard reference material HSST plate 01, Section 01MY) and will be irradiated

*In the spring of 1979, the staff was notified by Combustion Engineering that fluence calculations for CE reactor vessels may be incorrect (too low). Combustion Engineering is currently recalculating the fluence values. Also, the staff is having Brookhaven National Laboratory check these calculations.

along with the surveillance test specimens. The standard reference material was obtained through Subcommittee II of ASTM Committee E-10 on Radioisotopes and Radiation Effects. Use of standard reference material test specimens permits correlation of the post-irradiation data obtained in the course of this surveillance program with data obtained from other surveillance programs or irradiation experiments. In addition, changes in impact properties of the correlation monitors will provide a cross check on the neutron dosimetry.

Two sets of flux monitors are installed in each capsule. The first set consists of U^{238}, S^{32}, Fe^{54}, Ni^{58}, Cu^{63} and Ti^{46} monitors that will determine the neutron spectrum. The second set of monitors is composed of iron wires in stainless steel sheaths and will be used to measure the flux attenuation through the thickness of the Charpy specimen.

Unirradiated specimens were tested by Battelle Columbus Laboratories and the results reported in Reference 4. These results will be used as baseline data in determining radiation damage. In addition to the specimen types described above, unirradiated drop weight specimens were included. These specimens, in conjunction with Charpy tests, are used to calculate the initial RT_{NDT} values of the vessel materials.

To date, one surveillance capsule has been pulled from the vessel and shipped to Battelle Columbus. Test results are expected to be available by early 1979.

Although this surveillance program was developed in accordance with ASTM E 185-66, our review shows that it meets and in many areas exceeds the requirements of ASTM E 185-73. This is possible because of the many extra ingredients included in the program such as the number of capsules, Charpy specimens in both longitudinal and transverse directions and the number of Charpy specimens. Thus, this program is in accordance with Appendix H, 10 CFR Part 50, and will provide an excellent basis for predicting radiation damage on the vessel materials throughout their service life.

Pressure-Temperature Operating Limits

In a letter dated January 6, 1977, Consumers Power Company requested a change to Technical Specification 3.1.3 of Palisades regarding pressure-temperature operating limits (Ref. 5). The proposed operating limits were calculated in accordance with Appendix G, 10 CFR Part 50, and Appendix G to ASME Code Section III. The limiting material for these limits was base metal having an initial RT_{NDT} of -30°F. The limits were proposed for operation through 3×10^6 MWD_t. The NRC staff reviewed this proposal and concluded that additional tests would be required to verify an RT_{NDT} value of -30°F. Thus, the limits were approved but for a reduced operating period of 2.2×10^6 MWD_t. On February 22, 1978, Consumers Power Company submitted the results of mechanical tests on unirradiated vessel specimens and revised pressure-temperature operating curves (Refs. 4,7). These operating curves are based on an initial RT_{NDT} of 0°F and are proposed for operation through 3×10^6 MWD_t. From the submitted data on unirradiated specimens, we concluded that the limiting material had an initial RT_{NDT} value of 0°F and a copper content of 0.25%. Radiation damage, the increase in RT_{NDT}, was estimated from Regulatory Guide 1.99, Revision 1. These operating limits were found to be in accordance with Appendix G and were incorporated into the Technical Specifications as Amendment No. 41 (Ref. 8). Prior to the expiration of these operating limits, data on irradiated specimens will be available to provide the basis for new limits.

Inservice Inspection Program

Selected components of the Palisades nuclear reactor primary coolant system, including the reactor vessel, were subjected to a preoperational nondestructive examination in 1969 and 1970. Examinations were performed by Southwest Research Institute personnel in accordance with the rules of Appendix IX, ASME Code Section III (Ref. 9). Mechanized and manual ultrasonic techniques, and visual and liquid penetrant methods were used. The mechanized equipment used to examine the vessel (from the ID) was not intended to be used for later inservice inspections. In April 1971, a different remotely operated positioning mechanism was successfully used to perform a preservice examination of the vessel from the ID. This device is capable of being used for inservice examinations. No cracks were detected in these examinations. The only indications found were due to geometry, such as indications in closure head flange weld, longitudinal welds in the upper shell and lower shell, and on some nozzle to shell welds. The results of this examination will provide a baseline for future inservice inspections.

Inservice examinations on reactor vessel components were conducted by Southwest Research Institute personnel in 1973, 1976 and 1978 (Ref. 10). These examinations were performed in accordance with the 1971 Edition of ASME Code Section XI. The only reportable indications detected were on the closure washers. These were removed by grinding. The first inspection interval is scheduled to end on June 30, 1983.

Consumers Power Company, in letter dated May 3, 1977 with additional information supplied on October 7, 1977, submitted a revised inservice inspection program in accordance with 10 CFR 50.55a(g). This program is in conformance with ASME Code Section XI, 1974 Edition and Addenda through Summer 1975 (Ref. 11). This program will govern inspections for the second inspection period, October 31, 1976, to February 28, 1980. The NRC technical staff has completed their review of the Class 1 inspection program and concluded that it is acceptable (Ref. 12). One request for relief was granted; examination of nozzle to safe end welds in accordance with Category B-F requirements. The staff agreed to this request because the safe ends between the carbon steel nozzles and carbon steel piping are also made of carbon steel. Thus, these are not dissimilar metal safe ends. The staff will require that these safe end welds be examined to Category B-J requirements with the added requirement that the inspection be expanded to include 100% of each weld.

Consumers Power Company submitted a further revision to the Palisades inservice inspection program in letter dated June 13, 1978 (Ref. 13). This program contains requests for relief of some Category B-B, B-C and B-D inspection requirements. The staff has completed its review of the proposed inspection program. The required changes to the Technical Specifications, including the related Safety Evaluation, were made by Amendment No. 53 dated October 15, 1979. Relief from Code inspection requirements was granted for several items such as: (1) volumetric examination of the circumferential weld in the closure head, and (2) delay examination of vessel to flange, head to flange, and inlet and outlet nozzle welds.

Generic Safety Items

Based on the unirradiated properties of the Palisades reactor vessel materials, the NRC staff concludes that the materials of this vessel will have acceptable Charpy upper-shelf energy (fracture

toughness) values throughout service life. However, we will rereview this evaluation when the data from the irradiated surveillance capsule removed from the vessel in 1978 becomes available.*

The other generic item applicable to Palisades is overpressurization protection. Palisades has reported one overpressurization incident that occurred on September 1, 1974 while performing a leak test (Ref. 14). In August 1976, NRC requested that Consumers Power Company take steps, including design modifications, necessary to preclude exceeding the limits of Appendix G, 10 CFR Part 50, during inadvertent pressure transients (Ref. 15). Over the next year, Consumers Power Company submitted documentation in support of the overpressurization protection system (OPS) proposed for Palisades (Refs. 16, 17). Low temperature reactor vessel overpressure protection is provided by two pressurizer power operated relief valves (PORV) and two shutdown cooling system safety valves (SDC). The PORVS are solenoid-operated valves that have a high pressure setpoint (around 2400 psig) during normal operation. The PORV actuation circuitry has been modified so that a low pressure setpoint, 415 psig, is manually enabled at RCS temperatures below 300°F, and the two SDC safety valves are connected to RCS for temperature below 225°F. A pressure transient caused by mass addition (HPSIP or charging pump) or heat addition (decay heat, pressurizer heaters or starting a reactor coolant pump with a hot steam generator) is terminated below the Appendix G limits by automatic operation of these valves. The staff has completed its review of the Palisades OPS and has found it acceptable. The Technical Specifications have been revised to include testing and administrative procedures recommended by the staff. These specifications, including the related Safety Evaluation, were incorporated into the Technical Specifications by Amendment No. 31 dated September 28, 1979.

Conclusions and Recommendations

The Palisades reactor vessel was designed to the 1965 Edition of ASME Code Section III. The requirements of this Code were supplemented by Combustion Engineering specifications that required additional quality control measures. To further verify the initial integrity of the reactor vessel, a preservice nondestructive examination was performed. From the review of the design criteria and the results of the preservice examination and the fracture toughness tests on the vessel materials, it is concluded that the initial integrity of the vessel is acceptable. Three inservice examinations on reactor vessel components have been performed. To date, only very minor unacceptable indications have been detected. The effect of irradiation on the vessel's beltline materials is monitored by the Palisades material surveillance program. This program has been reviewed and is considered acceptable. It not only meets the main requirements of Appendix H, 10 CFR Part 50, but in some areas actually exceeds the Appendix H requirements. The reactor vessel is currently operating with pressure-temperature limits that are in conformance with Appendix G, 10 CFR Part 50. The staff will continue to review and update these operating limits to account for additional radiation damage on the vessel beltline materials. The degree of this radiation damage will be determined from the test results on material surveillance specimens. To date one capsule has been removed from the vessel. NRC has not received the results of these tests. However, from the results of

*The NRC staff received a copy of the report on the test results of this capsule in May 1979. (Reference Battelle Columbus Laboratories Final Report dated March 13, 1979.) The surveillance specimens received a fluence of 4.4×10^{19} n/cm^2, which is higher than the vessel wall ID will see at 32 EFPY. The upper shelf energy of both plate and weld materials at this fluence was above 50 ft-lbs. The shift in RT_{NDT} of weld metal is on the upper limit line of Regulatory Guide 1.99.

tests on unirradiated specimens it appears that the fracture toughness properties of the vessel materials will be acceptable throughout the vessel's service life.* The combination of these inservice inspections, conservative operating limits and the use of materials having adequate fracture toughness properties, provides assurance that the vessel integrity will be maintained at acceptable levels throughout service life. The generic safety items applicable to Palisades (low upper-shelf toughness and overpressurization protection) have been successfully resolved and will not adversely affect the integrity of the reactor vessel.

The following recommendation is made:

1. In view of the error found in the fluence calculations for Fort Calhoun, we recommend that Consumers Power Company recalculate the fluence estimates for Palisades and submit these estimates, along with the procedures used, to NRC for our review.

<u>References (Appendix J)</u>

1. Palisades FSAR, through Rev. 12/15/73.**

2. Consumers Power letter (D. P. Hoffman) to Director, NRR, dated May 23, 1978.**

3. Combustion Engineering Report dated 1968 attached to Reference 2.**

4. J. Perrin and E. Fromm, Battelle Columbus Laboratories Report dated August 25, 1977. (Attached to Consumers Power Company letter, D. P. Hoffman to Director, NRR, dated February 22, 1978.)**

5. Consumers Power letter (D. A. Bixel) to Director, NRR, dated January 6, 1977.**

6. NRC letter (A. Schwencer) to Consumers Power Company dated May 5, 1977.**

7. Consumers Power Company letter (D. P. Hoffman) to Director, NRR, dated February 22, 1978.**

8. NRC letter (D. Ziemann) to Consumers Power Company dated May 16, 1978.**

9. Southwest Research Institute Report SwRI 17-2249, July 25, 1971.**

10. Southwest Research Institute Reports SwRI 17-3620-001 dated November 1974 and 17-3620-140 dated June 1976.**

11. Consumers Power Company letter (D. P. Hoffman) to NRC (ORB-1) dated May 3, 1977.**

*The NRC staff received a copy of the report on the test results of this capsule in May 1979. (Reference Battelle Columbus Laboratories Final Report dated March 13, 1979.) The surveillance specimens received a fluence of 4.4×10^{19} n/cm^2, which is higher than the vessel wall ID will see at 32 EFPY. The upper shelf energy of both plate and weld materials at this fluence was above 50 ft-lbs. The shift in RT_{NDT} of weld metal is on the upper limit line of Regulatory Guide 1.99.

**Available in NRC PDR for inspection and copying for a fee.

12. NRC internal memorandum, L. Shao, Chief, Engineering Branch, to ORB-2 dated May 8, 1978.**

13. Consumers Power Company Letter (D. Hoffman) to NRC (ORB-2) dated June 13, 1978.*

14. Abnormal Occurrence Report No. AO-18-74, Docket 50-255.*

15. NRC letter (A. Schwencer) to Consumers Power Company dated August 11, 1976.*

16. Combustion Engineering Owner's Group, "Generic Report Overpressure Protection for Operating CE NSSS," December 3, 1976.*

17. Consumers Power Company letters (D. P. Hoffman) to Director, NRR, dated March 8, 1977, June 24, 1977, November 28, 1977 and January 3, 1978.*

*Available in NRC PDR for inspection and copying for a fee.
**Available in source file for USNRC Report NUREG-0569.

APPENDIX K

LACROSSE BOILING WATER REACTOR
PRESSURE VESSEL

Design

The nuclear steam supply system for LaCrosse Boiling Water Reactor was designed by Allis-Chalmers Manufacturing Company. The reactor vessel was also designed and manufactured by Allis-Chalmers in accorddance with the rules of ASME Code Section VIII and Nuclear Code Case 1270N (Ref. 1). In addition to the minimum Code requirements, Allis-Chalmer's specifications required the following quality enhancing practices (Ref. 2):

1. The vessel stress analysis included analysis of thermal transient and fatigue effects. The method used was based upon the method of analysis developed for Naval Reactors. The method is given in PB-151987, "Tentative Structural Design Basis for Reactor Pressure Vessels and Associated Components." The vessel stress analysis performed to the procedures outlined in this document together with the code requirements is essentially equivalent to that required by ASME Section III for Class 1 vessels.

2. The vessel was constructed of SA-302, Grade B, plate and SA-336 forging material. These materials were Charpy V-notch impact tested. They are essentially equivalent to the SA-533, Grade B, Class 1, and SA-508, Class 2, materials being used today.

3. All forging material in the vessel pressure boundary was ultrasonically inspected.

4. All plate material in the vessel pressure boundary was ultrasonically inspected.

5. Stainless steel cladding in the vessel shell and head regions was ultrasonically inspected for bonding to the base metal.

6. Carbon steel pressure containing welds were magnetic particle inspected in accordance with the following:

   a. After each of the first two passes.
   b. When the weld was half completed.
   c. On the finished surface of the weld prior to any stress heat treatment.
   d. At the root of the weld after back chipping.
   e. After intermediate stress relief and prior to stainless steel cladding.

A tabulation of material acceptance inspections for the LaCrosse reactor vessel as compared to those required by ASME Sections VIII and III is included in Table K-1. The NSSS of LaCrosse is unique since it is one of the very few BWRS not designed by General Electric.

The reactor pressure vessel consists of a cylindrical, plate-welded shell section with a formed integral hemispherical bottom head and a removable hemishperical top head which is bolted to a mating flange on the vessel shell to provide for vessel closure. The vessel has an overall inside height of 37 ft, an inside diameter of 99 inches and a nominal wall thickness of 4 inches (including 3/16 inch of integrally-bonded stainless steel cladding).

The vessel design pressure is 1415 psig and its operating pressure is 1250 psig. Steam is withdrawn from the reactor vessel through two 8-inch steam lines leading to a single 10-inch steam line that passes from the containment shell to the turbine building. Condensate returns from the turbine building through an 8-inch feedwater line, which connects to the 16-inch forced-circulation pump suction manifold where the condensate is mixed with the recirculating coolant before entering the recirculation pumps.

The date of commercial operation is November 1, 1969. As of January 1, 1979, the plant has operated for approximately 5 EFPY. This vessel is designed for 20 EFPY. The maximum fluence on the vessel wall ID at end of life is estimated to be 3.2×10^{19} n/cm^2.

Materials

The cylindrical part of the vessel is composed of nine shell courses. These courses are made from SA-302, Grade B, plate material obtained from Lukens Steel Company. The plates are joined with one longitudinal seam weld in each shell course. The shell courses are joined by circumferential welds. All welds are made by the submerged arc process using Linde Oxweld 40 wire. All welded components were subjected to a final stress relief consisting of heating to 1100°F-1175°F, holding for 1 hour per inch of thickness, furnace cooling to 500°F, and air cooling to room temperature (Ref. 3).

Data from chemical analysis of vessel plate and weld material are not available. However, the chemistry of typical material can be obtained from surveillance specimens: three plate materials and one weld. The weld metal has a copper content of 0.18%. A chemical analysis was made on the plate materials but copper content was not determined. It is recommended that a complete chemical analysis be performed on material already tested or on the next batch of surveillance material removed from the reactor vessel.

For the three plates in the beltline region, the upper shelf energy varies from 60 to 120 ft-lbs and the highest RT_{NDT} is 30°F. For weld metal, the RT_{NDT} is estimated as -30°F and the upper shelf energy is 60 ft-lbs (values obtained from the surveillance weld metal). These values are slightly on the low side, but they are acceptable because of the small vessel size and the low vessel stresses.

Material Surveillance Program

The LaCrosse material surveillance program was developed and planned in 1966 (Ref. 4). The program consists of 20 capsules attached to the inner surface of the thermal shield at the midplane of the core The capsules are positioned circumferentially so that the lead factor for all capsules is approximately two. There are two types of capsules: A and B. All capsules contain 22 Charpy and 6 tensile specimens from vessel materials and 6 Charpy specimens from a correlation material. The 6 tensile specimens include 3 from plate NP 1055 and 3 from weld metal. Type A capsules contain 10 Charpy specimens from plate NP 1055, 6 from NP 1054, and 6 from weld metal. The Type B capsules contain 10 Charpy specimens from plate NP 1055, 6 from NP 1056 and 6 from weld metal.

TABLE K-1

MATERIAL AND ACCEPTANCE STANDARDS FOR LACROSSE

	ASME Section III		LaCrosse		ASME Section VIII	
Materials	Exam.*	Extent	Exam.*	Extent	Exam.*	Extent
Plates	U. T.	100% Volume	U. T.	100% Volume	N. S.	-
Forgings	U. T.	100% Volume	U. T.	100% Volume	N. S.	-
Fabrication						
Weld Grooves	M. T. or P. T.	100% Surface	M. T.	100% Surface	Visual	N. S.
Weld Joints	M. T. or P. T	100% Surface	M. T.	100% Surface	-	-
Shell and Head	R. T.	100% Volume	R. T.	100% Volume	R. T.	100% Volume
Nozzle Welds	R. T.	100% Volume	R. T.	100% Volume	R. T.	100% Volume
	M. T. or P. T.	100% Surface	M. T.	100% Surface	N. S.	-
Partial Penetration Welds						
Progressive	M. T. or P. T.	100% Surface	M. T. or P. T.	100% Surface**	Visual	N. S.
Final Surface	M. T. or P. T.	100% Surface	M. T. or P. T.	100% Surface	Visual	N. S.

*Examination Notations
R. T. - Radiography
U. T. - Ultrasonic Examination
M. T. - Magnetic Particle Examination
P. T. - Liquid Penetrant Examination
N. S. - Not specified.

**Performance of this examination could not be substantiated from review of documentation available at NRC. However, due to the extent of the progressive M.T. inspection performed on the pressure containing welds of the LaCrosse vessel, it is likely that it was performed.

Plates 1054 and 1055 are in the beltline region. Plate 1056 is located in the intermediate shell course and is above the beltline region. The correlation material was obtained from Chicago Operations Office of the Atomic Energy Commission and was taken from the ASTM A302B material reported on in Pacific Northwest Laboratory Report BNWL-CC-236, "Fabrication History of Alloys Used in the Irradiation Effects on Reactor Structural Materials Program." The weld material was obtained from a weld on plate NP 1056. All specimens from plate material were oriented in the longitudinal direction (strong) and machined from material at the 1/4 T location. Weld metal specimens were also machined from material at the 1/4 T location.

Three types of flux wires are contained in the surveillance capsules. These are 0.021-inch diameter pure iron wire, 0.020-inch diameter pure nickel wire and 0.020-inch diameter aluminum/0.1% cobalt wire. One piece of each wire, approximately 1½ inches long, is located in the V-notch area of each of the seven layers of Charpy specimens in the test specimen capsules. Two vessel wall flux wire assemblies, each containing iron, nickel and aluminum/0.1% cobalt wires, are located 180° apart in the annulus between the vessel wall and the thermal shield. These assemblies, which extend the length of the core, are also fitted with a cable and lifting ring.

Since the program was initiated prior to the issuance of Appendix H, 10 CFR Part 50, it does not completely meet all the Appendix H requirements. However, there are many areas in which this program exceeds Appendix H requirements. The main areas where this program differs from Appendix H are the number of capsules, no HAZ material specimens and the specimen orientation. By decreasing the Charpy energy of longitudinally oriented specimens by 65% we obtain values that can be conserva tively compared to transverse orientated specimens. It is also noted that this program contains no specimens from HAZ material. However, since test results on HAZ material from other surveillance programs show that this material has good resistance to radiation damage and is not a vessel limiting material, we conclude that this exception does not limit the effectiveness of the program. The number of Charpy specimens varies from 6 to 10 for various materials in the program. However, it is noted that there are four times the Appendix H required number of capsules. Since the withdrawal schedule calls for capsule removal in pairs, the effective number of specimens doubles and thus actually exceeds the 12 required by Appendix H. Also, on the plus side, this program contains a monitor material and samples from three different vessel plate materials; all of which is above Appendix H requirements. We therefore conclude that the LaCrosse surveillance program is acceptable for predicting radiation damage on vessel materials throughout service life.

To date two capsules were removed from LaCrosse during the 1972 refueling outage and four were removed during the 1975 refueling outage. The average fast fluence received by the two capsules removed in 1972 is 4.35×10^{18} n/cm^2. Test results show that weld metal has an increase in RT_{NDT} of 85°F at 30 ft-lbs. This radiation also dropped the upper shelf energy from about 75 to 70 ft-lbs. Perhaps the most interesting data was the large drop in upper shelf energy of plate NP 1054, from over 100 to about 50 ft-lbs. The irradiated upper shelf values for the other two plate materials were about 100 ft-lbs.

The neutron fluence on the four capsules withdrawn from the vessel in 1975 is 6.6×10^{18} n/cm^2 on capsules 9A and 9B, and 1.1×10^{19} n/cm^2 on capsules 2A and 7B (Ref. 6). Material from plate NP 1054 was included in capsules 9A and 2A. From these tests it showed an upper shelf energy in excess of 100 ft-lbs with five data points. This data is considered valid. No reason for the unusually large drop in shelf energy obtained in the earlier test series can be found. The staff will continue to watch the test results on this material obtained in future surveillance tests for signs of low

upper shelf energy. Based on test results so far, we conclude that there is no low upper shelf problem with this material. Weld metal from capsules 2A and 7B had an increase in RT_{NDT} based on lateral expansion of 115°F and an upper shelf energy of about 60 ft-lbs. Weld material in capsules 9A and 9B had an upper shelf energy of almost 75 ft-lbs. These values of shelf energy are well above the requirements of Appendix G, 10 CFR Part 50.

The next capsule is scheduled for removal at 6 EFPY.

Pressure-Temperature Operating Limits

The original pressure-temperature operating limits for LaCrosse were based on a criteria using RT_{NDT} + 60°F. Upon the initial issuance of Appendix G, 10 CFR Part 50, new operating limits were required. By letter dated August 3, 1976, Dairyland Power Cooperative submitted revised operating limits based on the requirements of Appendix G (Ref. 7). The revised limit curves are calculated for an RT_{NDT} value of 0°F. These curves are required by the Technical Specifications to be increased by values of RT_{NDT} that include radiation damage. A curve for RT_{NDT} versus fluence is included in the Technical Specifications. The curve submitted in 1976 was based on the test results of the first two capsules removed from the vessel. During the first few years of operation, base metal is the limiting material. Later in life weld metal becomes the limiting material. NRC approved these operating limits as License Amendment 8 (Ref. 8). Following the completion and review of tests on the next four capsules removed from the vessel, it was determined necessary to revise the damage prediction curve in the Technical Specifications to provide for slightly more damage than originally anticipated. This revised damage estimate curve was reviewed and approved by NRC and was issued as License Amendment 13 (Refs. 9, 10). The present pressure-temperature operating limits for LaCrosse are acceptable and are in accordance with Appendix G, 10 CFR Part 50. The damage prediction curve will be re-reviewed when the results of tests on the next material surveillance capsule are reviewed.

Inservice Inspection Program

In 1970 an inservice inspection program based on and generally conforming to the rules of ASME Code Section XI was initiated for LaCrosse. Since the reactor vessel was designed prior to the initial issuance of Section XI, the examination of some welds required by Section XI is not practical because of limited access. The main areas of nonconformance are the Category B-A welds in the reactor vessel beltline region and some of the Category B-B welds in the vessel. However, to partially compensate for not inspecting the above welds, the highly stressed vessel to flange and head to flange welds were scheduled for inspection at twice the Section XI required frequency. This is considered a positive contribution to the program. However, even though these welds are highly stressed, they do not receive any significant amount of radiation while the beltline welds are degraded by high irradiation.

To date, inspections on reactor vessel welds in accordance with Section XI rules have been conducted in 1971, 1973 and 1977 (Ref. 11). Items examined in these inspections include two thirds of the upper shell to intermediate shell weld, 100% of the vessel to flange and head to flange circumferential welds, a longitudinal weld, primary vessel nozzle and safe end welds, and vessel closure nuts and studs. No indications greater than those allowed by Section XI were reported. However, a series of laminar inclusions was detected in the vessel wall adjacent to the number 1 recirculation nozzle. This area has been reexamined six times and no growth has been noted.

The first inspection interval is scheduled to be completed in October 1979. Inspections are scheduled in the fall of 1978 and 1979 to complete this inspection interval. The fall 1978 inspection has been completed but the report of these examinations has not been reviewed as yet.

In accordance with the new provisions of paragraph 50.55a(g), 10 CFR Part 50, LaCrosse will commence its second inspection interval on September 13, 1979. A revised inspection program in accordance with later editions of Section XI is required for this inspection interval. Dairyland Power Cooperative is now planning this program and will submit it to NRC for review in 1979.

Generic Safety Items

The generic safety items reviewed for LaCrosse are low upper shelf toughness and sensitized stainless steel safe ends. Because of the design of the LaCrosse primary coolant system, the feedwater nozzle and control rod drive return line nozzle problems are not applicable to LaCrosse. The CRD excess flow is not returned directly to the reactor vessel but is discharged into the decay heat system. Hence, there is no CRD RL nozzle on the reactor vessel. Regarding the feedwater nozzle cracking problem, condensate returning from the condenser connects to the forced circulation pump suction manifold where it is mixed with the recirculation coolant. Thus, cold feedwater does not enter the reactor vessel. The temperature of the recirculating coolant is 563.5°F at normal operating conditions. This temperature is only slightly lower than the reactor vessel wall temperature so that thermal stresses will be low. During startup and shutdown transients, the inlet coolant temperature is also very close to the vessel wall temperatures. It should also be noted that the plant nomenclature refers to the inlet nozzles to the reactor vessel as recirculation nozzles and the forced circulation pump suction manifold nozzles as feedwater nozzles. To date no cracks have been detected in any of these nozzles.

Based on the results of surveillance specimens, we conclude that the LaCrosse reactor vessel will not have a low upper shelf energy problem. Tests have been completed on surveillance specimens made of weld and plate material at fluences up to 1.1×10^{19} n/cm^2. At this fluence level plate metal has an upper shelf energy of about 100 ft-lbs in the strong direction (referred to the weak direction this is about 65 ft-lbs). At this fluence weld metal showed a shelf energy of 60 ft-lbs. These values are well above the 50 ft-lbs required by Appendix G, 10 CFR Part 50.

The pressure vessel and piping system of LaCrosse incorporated Type 304 stainless steel safe-ends on several nozzles. The fabrication sequence for the vessel and recirculation piping was such that the various safe-ends were subjected to stress-relieving treatments at 1150°F for time periods ranging from 1¼ to 5 hours (Ref. 12). This treatment resulted in sensitization of the safe ends with a corresponding increase in sensitivity to intergranular corrosion and stress-corrosion cracking. Because of cracking in sensitized stainless steel safe ends reported in other operating reactors, an investigation of the degree of sensitization and the serviceability for continued operation of the LaCrosse safe ends was made by Southwest Research Institute. This study led to the replacement of safe ends on the following systems that connect directly to the reactor vessel: main steam, purification line, emergency cooling and boron injection (Ref. 13). Safe ends not directly connecting to the reactor vessel were also replaced, such as the feedwater nozzle and decay heat nozzle safe ends. From this present review we conclude that the LaCrosse reactor vessel does not have any sensitized stainless steel safe ends attached to the reactor vessel.

During the SEP review of LaCrosse, several items that relate to the integrity of the vessel but not covered in the specific topic of review were noted. The first item concerns a laminar indication found near the number 1 recirculation nozzle during the 1970 inservice inspection (Ref. 14). This area was reexamined by ultrasonic techniques in 1971 and 1973 and no indication of flaw growth was detected. It is believed that this flaw was in the vessel prior to final fabrication. The second item noted was the failure of two vessel closure studs in 1971 (Ref. 15). The failure of these studs was attributed to stress corrosion cracking in the thread roots. All closure studs have been replaced with SA 540, Grade B23 C14, material studs. This material has good resistance to stress corrosion. To date no further cracking has been detected in the LaCrosse closure studs.

Recommendations and Conclusions

The LaCrosse reactor vessel was designed to ASME Code Section VIII. However, the requirements of this Code were supplemented by the requirements of Nuclear Code Case 1270N, the Navy Code and purchase specifications so that the design and quality control measures utilized were essentially in accordance with the rules of ASME Code Section III. Therefore, the initial integrity of this vessel is considered acceptable. The primary stresses in the vessel beltline region are low, approximately 70% of those permitted by Section III. These low stresses along with the use of materials with adequate fracture toughness properties provide assurance that brittle fracture will not occur. Inservice inspections in accordance with ASME Code Section XI rules have been performed on reactor vessel components since 1971. The only significant indication detected in inservice inspections was a laminar flaw near the number 1 recirculation nozzle. Later examinations indicate that this flaw is not growing larger. The staff will require that inservice inspections be continued throughout the vessel's service life in accordance with Section XI rules. The reactor vessel is currently operating with pressure-temperature operating limits that are in accordance with Appendix G, 10 CFR Part 50. The staff will continue to review and update these operating limits to account for radiation damage on vessel materials. The amount of radiation damage will be determined from the results of the vessel's material surveillance program. The material surveillance program for LaCrosse has been reviewed and is considered acceptable. The combination of these inservice inspections, conservative operating limits, low vessel stresses and the use of materials having adequate fracture toughness properties provides assurance that the integrity of the vessel will be maintained at acceptable levels throughout service life. The generic safety items applicable to LaCrosse (low upper shelf toughness and sensitized stainless steel safe ends) have been successfully resolved and will not adversely affect the integrity of the vessel.

The following recommendations are made:

1. The staff should review the inservice inspection results for the 1978 and 1979 examinations and verify that there has been no growth in the laminar flaw detected in the 1970 inservice inspection. The staff should use these results as a bases for future examination requirements on this nozzle; i.e., whether or not to require an augmented examination schedule.

2. The reported chemical composition of materials in the material surveillance program is incomplete. A complete chemical analysis of surveillance materials should be performed on the next batch of materials removed from the reactor vessel or on materials that have already been tested.

References (Appendix K)*

1. LaCrosse Safeguards Report for Operating Authorization, August 1967, Section 4.

2. Allis-Chalmers Report ACNP-65534, May 10, 1965.

3. Dairyland Power Cooperative letter (J. Madgett) to Director, NRR, dated December 12, 1977.

4. Allis-Chalmers Report, ACNP 66513, February 1964.

5. E. B. Norris, Southwest Research Institute, Topical Report SwRI 02-3467, March 23, 1974.

6. E. B. Norris, Southwest Research Report, SwRI 02-4074-001, April 26, 1977.

7. Dairyland Power Cooperative letter (J. Madgett) to Director, NRR, dated August 3, 1976.

8. NRC letter (R. Reid) to DPC dated January 19, 1977.

9. Dairyland Power Cooperative letter (J. Madgett) to Director, NRR, dated January 11, 1978.

10. NRC letter (D. Ziemann) to DPC dated June 12, 1978.

11. Southwest Research Institute Reports SwRI 17-4835, December 1977; 17-2294-023, March 1974; and 17-2294-12, April 22, 1971.

12. Southwest Research Institute Reports SwRI 17-2757-09, November 13, 1970; and 17-1403-01-0001, November 12, 1969.

13. United Nuclear Corporation Report UNC-SS-561, April 10, 1970.

14. Southwest Research Institute Reports SwRI 17-2294-12, April 29, 1970; 17-2294-14, September 16, 1971; 17-2294-021, September 1973.

15. Southwest Research Institute Reports SwRI 2154-20, December 1971; 17-2294-17, October 19, 1973.

*All references in this Appendix are available in NRC PDR for inspection and copying for a fee.

APPENDIX L

SUMMARY OF DESIGN AND MATERIALS DATA

Following is a tabular summary of design and material data for nuclear reactor vessels reviewed under the Systematic Evaluation Program.

TABLE L-1

DESIGN AND MATERIAL DATA FOR SEP PLANTS

PLANT (DOCKET NO.)	TYPE	VESSEL DESIGN CODE	VESSEL SHELL MATERIAL
Dresden 1 (50-10)	BWR	Section I	SA-302, Grade B
Yankee Rowe (50-29)	PWR	Section VIII	SA-302, Grade B (Modified)
Big Rock Point (50-155)	BWR	Sections I and VIII, Navy, and Code Cases 1270N, 1271N and 1273N	SA-302, Grade B
San Onofre 1 (50-206)	PWR	Section VIII, Navy, and Code Cases 1270N and 1273N	SA-302, Grade B
Connecticut Yankee (50-213)	PWR	Section VIII, Navy, and Code Cases 1270N and 1273N	SA-302, Grade B
Oyster Creek (50-219)	BWR	Sections I and VIII, Navy, and Code Cases 1270N and 1273N	SA-302, Grade B (Modified)
LaCrosse (50-409)	BWR	Section VIII, Navy, and Code Case 1270N	SA-302, Grade B
Ginna (50-244)	PWR	Section III, 1965 Edition	SA-508, Class 2
Millstone 1 (50-245)	BWR	Section III, 1965 Edition	SA-302, Grade B (Modified)
Dresden 2 (50-247)	BWR	Section III, 1963 Edition	SA-302, Grade B (Modified)
Palisades (50-255)	PWR	Section III, 1965 Edition	SA-302, Grade B (Modified)

APPENDIX M

DISCUSSION OF ASME CODE, SECTIONS I AND VIII

These codes were written for nonnuclear vessels which are used in fossil fuel power generating plants under service conditions similar to those nuclear vessels are exposed to. The service record of such vessels has been excellent. A discussion of the operating experience of nonnuclear vessels designed to these codes and a tabulation of the differences between ASME Sections I and III can be found in WASH-1318, "Analysis of Pressure Vessel Statistics from Fossil-Fueled Power Plant Service and Assessment of Reactor Vessel Reliability in Nuclear Power Plant Service." Except as noted, the requirements of Section VIII are basically the same for the types of vessels discussed herein.

The more significant requirements of Sections I and VIII can be summarized as follows:

1. These codes gave formulas for calculating average stresses in the vessel and assigned conservative allowable stress levels that had sufficient margin to cover localized stress effects which the codes did not require to be calculated. Analyses for stress effects due to transient loads and for fatigue effects resulting from cyclic loads over the service life of the vessel were not required.

   The allowable stress levels for reactor vessel pressure boundary materials were about 25% lower than would be permitted by ASME Section III for similar materials. Accordingly, vessels designed to Section I and VIII generally have heavier wall thicknesses than would be required by ASME Section III.

2. Detailed inspection requirements for vessel materials were not specified and were generally left up to the discretion of the vessel manufacturer. An exception to this is the examination of welds. Sections I and VIII require 100% radiography of all full penetration welds.

3. In addition to radiography, weld procedures and welders were qualified prior to production welding. Sample weld assemblies were made, and the quality of the weld verified by both destructive and nondestructive test methods.

4. Overpressure protection requirements as applied for nuclear reactor vessels were more stringent than now required by ASME Section III. Total pressure relieving capacity was computed with no credit being given for the ameliorating effect of shutting the reactor down, i.e., reactor scram. Section III permits taking credit for the effect of the scram and thus theoretically requires less total pressure relieving capacity for the same service conditions. Maximum pressures permitted under Sections I and VIII for anticipated pressure surge conditions were, respectively, 6% above design pressure for Section I and 10% above design pressure for Section VIII; Section III permits 10%.

5. These codes require that a vessel be hydrostatically tested before being placed into service, at a test pressure of 1.5 times the vessel design pressure. The same test under ASME Section III is performed at 1.25 times the design pressure.

APPENDIX N

DISCUSSION OF THE NAVY CODE AND NUCLEAR CODE CASES

Navy Code (PB-151987), "Tentative Structural Design Basis for Reactor Pressure Vessels and Directly Associated Components

The Navy Code was a forerunner of the design section of ASME Section III. It was written primarily for application to the design of reactor pressure vessels used in the Naval Reactor Program. It contained essentially all of the requirements for detailed vessel stress analyses that subsequently appeared in Section III. Like ASME Section III, it provided for the following:

1. It utilized the maximum shear stress theory of failure instead of the maximum stress theory.

2. It required a detailed calculation and classification of all stresses and applied different stress limits to classes of stresses.

3. It required a detailed fatigue analysis and provided rules for prevention of fatigue failure.

ASME Nuclear Code Cases

1270N - Provided the following "General Requirements" for nuclear vessels:

1. That the vessels be constructed to ASME Section I or VIII modified by the requirements of the Nuclear Code Cases.

2. Established order of precedence between possible conflicting requirements of the Nuclear Code Cases and ASME Sections I and VIII, i.e., Nuclear Case requirements were to have precedence.

3. That vessel purchase specifications include additional requirements to ensure vessel integrity in the unique nuclear environment for the intended life of the vessel.

1271N - This case provided guidance on selection and utilization of pressure-relieving devices for use in conjunction with a radioactive working fluid. In addition, it recommended that a minimum of two such devices be utilized.

1273N - Significant requirements of this case include the following:

1. Steady-state thermal stresses were to be combined with primary and secondary stresses resulting from the design pressure. The resulting combination of stresses was limited to three times the allowable stress S at the design temperature (similar to procedure used in ASME Section III).

2. Maximum allowable design stresses for bolting material, operating at temperatures up to 800°F, were limited to one-third the material yield strength at temperature (similar to ASME Section III).

3. Compensation, i.e., additional reinforcement, was required for all openings regardless of diameter. Compensation was to be determined on either the reinforcement or the ligament efficiency basis as defined in ASME Sections I or VIII.

4. Detailed design and inspection requirements were provided for both full and partial penetration pressure boundary welds. Similar requirements were ultimately written in to ASME Section III.

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16. ABSTRACT (*200 words or less*)

This report includes a documented review of the integrity of the eleven reactor pressure vessels covered in the Systematic Evaluation Program. This review deals primarily with the design specifications and quality assurance programs used in the vessel construction and the status of material surveillance programs, pressure-temperature operating limits, and inservice inspection of the applicable plants.

Several generic items such as PWR overpressurization protection and BWR nozzle and safe-end cracking also are evaluated. The eleven vessels evaluated include Dresden Units 1 and 2, Big Rock Point, Haddam Neck, Yankee Rowe, Oyster Creek, San Onofre, LaCrosse, Ginna, Millstone 1, and Palisades.

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