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RANCHO SECO NUCLEAR GENERATING STATION
UNIT NUMBER 1
INSERVICE INSPECTION PROGRAM
TECHNICAL EVALUATION REPORT

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Science Applications, Inc.
McLean, Virginia 22102

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Science Applications, Inc.

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TECHNICAL EVALUATION REPORT
RANCHO SECO NUCLEAR GENERATING STATION
UNIT NO. 1
INSERVICE INSPECTION PROGRAM

INTRODUCTION

The revision to 10 CFR 50.55a, published in February 1976, required that Inservice Inspection (ISI) Programs be updated to meet the requirements (to the extent practical) of the Edition and Addenda of Section XI of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code* incorporated in the Regulation by reference in paragraph (b). This updating of the programs was required to be done every 40 months to reflect the new requirements of the later edition of Section XI.

As specified in the February 1976 revision, for plants with Operating Licenses issued prior to March 1, 1976, the regulations became effective after September 1, 1976, at the start of the next regular 40-month inspection period. The initial inservice examinations conducted during the first 40-month period were to comply with the requirements in editions of Section XI and addenda in effect no more than six months prior to the date of start of facility commercial operation.

The Regulation recognized that the requirements of the later editions and addenda of the Section XI might not be practical to implement at facilities because of limitations of design, geometry, and materials of construction of components and systems. It therefore permitted determinations of impractical examination or testing requirements to be evaluated. Relief from these requirements could be granted provided health and safety of the public were not endangered giving due consideration to the burden placed on the licensee if the requirements were imposed. This report provides evaluations of the various requests for relief by the licensee, Sacramento Municipal Utility District (SMUD), of the Rancho Seco Unit No. 1. It deals only with inservice examinations of components and with system pressure tests. Inservice tests of pumps and valves (IST programs) are being evaluated separately.

* Hereinafter referred to as Section XI or Code.



The revision to 10 CFR 50.55a, effective November 1, 1979, modified the time interval for updating ISI programs and incorporated by reference a later edition and addenda of Section XI. The updating intervals were extended from 40 months to 120 months to be consistent with intervals as defined in Section XI.

For plants with Operating Licenses issued prior to March 1, 1976, the provisions of the November 1, 1979, revision are effective after September 1, 1976, at the start of the next one-third of the 120-month interval. During the one-third of an interval and throughout the remainder of the interval, inservice examinations shall comply with the latest edition and addenda of Section XI, incorporated by reference in the Regulation, on the date 12 months prior to the start of that one-third of an interval. For Rancho Seco, the ISI program and the relief requests evaluated in this report covers the second and third period of the first interval, i.e., from October 18, 1979, to June 18, 1986. This program was based upon the 1974 Edition of Section XI of the ASME Boiler and Pressure Vessel Code with Addenda through the Summer of 1975.

The November 1979 revision of the Regulation also provides that ISI programs may meet the requirements of subsequent code editions and addenda, incorporated by reference in paragraph (b) and subject to Nuclear Regulatory Commission (NRC) approval. Portions of such editions or addenda may be used provided that all related requirements of the respective editions or addenda are met. These instances are addressed on a case-by-case basis in the body of this report.

Finally, Section XI of the Code provides for certain components and systems to be exempted from its requirements. In some instances, these exemptions are not acceptable to NRC or are only acceptable with restrictions. As appropriate, these instances are also discussed in this report.

References (1) to (15) listed at the end of this report pertain to previous information transmittals on ISI between the licensee and the NRC. By letter of April 22, 1976,⁽¹⁾ the Commission provided general guidance to the licensee, and the licensee responded to that guidance on May 19, 1976.⁽²⁾ The Commission issued further generic guidance on November 17, 1976.⁽³⁾ On January 24, 1978,⁽⁴⁾ the licensee requested an extension on his commitment made in response to the initial guidance. In its response on February 15, 1978,⁽⁵⁾ the Commission did not grant an extension. On February 27, 1978,⁽⁶⁾ the licensee asked that additional outage time be added to the ISI interval, requesting an October 18, 1979, end date. The Commission granted this request on May 30, 1978.⁽⁷⁾ A proposed Amendment 63 containing Technical Specifications



changes was submitted by the licensee on March 16, 1979,⁽⁸⁾ and he submitted relief requests on July 18, 1979.⁽⁹⁾ Questions on these submittals were raised at meetings on October 3 and 4, 1979,⁽¹⁰⁾ of NRC and SMUD personnel. On October 17, 1979,⁽¹¹⁾ the NRC granted interim relief based on the licensee's July submittal and formally requested answers to the questions raised in the October meeting. The questions, which were not formally transmitted, were answered by letter by the licensee on December 10, 1979,⁽¹²⁾ and on December 12, 1979, the licensee submitted a revision to the proposed Amendment 63.⁽¹³⁾

By letter of March 15, 1982,⁽¹⁴⁾ the NRC requested additional information to complete its ISI review. This information was furnished by the licensee on April 19, 1982.⁽¹⁵⁾

From these submittals, a total of 13 requests for relief from Code requirements or for updating to a later Code were identified. These requests are evaluated in the following sections of this report.



I. CLASS 1 COMPONENTS

A. Reactor Vessel

1. Relief Request NDE-1, Vessel-to-Flange Circumferential Weld, Category B-C, Item B1.3

Code Requirement

Category B-C of Table IWB-2500 requires that 100% of each circumferential weld be volumetrically examined each inspection interval. Paragraph IWB-2411 further stipulates that at least 25% but no more than 33-1/3% of the required examinations be complete within the first one-third of an interval; at least 50% but no more than 56-2/3% be complete within the second one-third of an interval; and that the remainder of the examinations be completed by the end of the inspection interval.

Code Relief Request

Relief is requested from the schedule in Paragraph IWB-2411 for the flange-to-upper shell seam weld.

Proposed Alternative Examination

During the first one-third interval, all accessible areas of the weld will be examined from the top or flange surface by utilizing a full 360° sweep of the vessel circumference. During the second one-third interval, any recordable indications which are noted during the first inspection will be reinspected manually. One hundred percent of the weld will be examined from the vessel ID in the final one-third interval.

Licensee's Basis for Requesting Relief

The reactor vessel flange-to-upper shell course weld is not accessible from the vessel inside diameter (ID) unless the core support assembly has been removed. Further, only approximately 50% of the weld is accessible from the top or flange mating surface due to stud holes and placement of the automated reactor inspection tool.

Evaluation

The licensee has agreed to the following examination schedule:

1st Period: (1/3 interval) - 100% of the accessible portion of the weld is to be examined from the flange surface.



2nd Period: (2/3 interval) - Manual examination of any recordable indications found during the first period.

3rd Period: (End of 10-year interval) - 100% of weld examined from vessel ID.

To conform to Code, the licensee would have to remove the core support assembly each period, which is not practical. This is recognized in the more recent version of the Code (1977 Edition through Summer 1978 Addenda) in which deferral of the examination is allowed.

The 1977 Edition of Section XI has been referenced in 10 CFR 50.55a, and inservice examinations may meet the requirements of this edition in lieu of those from previous editions with the following provisions:

- (a) Commission approval is required to update to the more recent edition (pursuant to 10 CFR 50.55a(g)(4)(iv));
- (b) When applying the 1977 Edition, all of the addenda through Summer 1978 Addenda must be used;
- (c) Any requirement of the more recent edition which is related to the one(s) under consideration must also be met.

Conclusions and Recommendations

Based on the above evaluation, relief from Code requirements on volumetric examination should not be granted. Instead, pursuant to 10 CFR 50.55a(g)(4)(iv), approval should be granted to update to the requirements of the 1977 Edition, Summer 1978 Addenda for Category C-D items. This approval would allow deferral of the examination to the end of the inspection interval.

References

Reference 9 (p 1 and NDE-1).



2. Relief Request NDE-2, Primary Nozzle-to-Vessel Welds and Nozzle Inside Radiused Section, Category B-D, Item B1.4

Code Requirement

Category B-D of Table IWB-2500 requires that 100% of nozzles be volumetrically examined each inspection interval. Paragraph IWB-2411 further stipulates that at least 25% but no more than 33-1/3% of the required examinations be complete within the first one-third of an interval; at least 50% but no more than 66-2/3% be complete within the second one-third of an interval; and that the remainder of the examinations be completed by the end of the inspection interval.

Code Relief Request

Relief is requested from the schedule in Paragraph IWB-2411.

Proposed Alternative Examination

Two (2) outlet nozzles will be inspected from the nozzle bore only during the first one-third interval. All nozzles will be inspected from their bores and the reactor vessel ID during the final one-third interval.

Licensee's Basis for Requesting Relief

Six (6) of eight (8) reactor vessel nozzles are inaccessible without removal of the core support assembly. Therefore, examination of two (2) inlet nozzles will be deferred from the second to the final one-third interval.

Evaluation

To conform with Code, the licensee would have to remove the core support assembly during both the second and third periods, which is not practical. The increase in personnel exposure is not warranted by the marginal increase in safety. This is recognized in the Code for the reactor vessel Category B-A and B-B welds which are covered by the core support assembly and are only examined at or near the end of the interval. The schedule proposed by the licensee is acceptable. The total volume of weld examined during the 10-year interval exceeds Code requirements since the outlet nozzles will be examined twice.

Conclusions and Recommendations

Based on the above evaluation, it is concluded that for the welds discussed above, the code requirements are impractical. It is further concluded that the alternative examination discussed



above will provide necessary added assurance of structural reliability.

Therefore, it is recommended that relief from the schedule given in Paragraph IWB-2411 be granted, provided the schedule proposed by the licensee is adhered to.

References

Reference 9 (p 2 and NDE-2).



3. Relief Request NDE-3, Pressure Retaining Bolting,
Category B-G-2, Item B1.11

Code Requirement

Visual examination of 100% of the bolts, studs and nuts each interval is required. Bolting may be examined either in place under tension, when the connection is disassembled, or when bolting is removed.

Code Relief Request

Relief is requested from examining 100% of the Control Rod Drive Mechanism (CRDM) bolts and housing flange rings.

Proposed Alternative Examination

Only bolting from Control Rod Drive mechanisms that are removed will be examined.

Licensee's Basis for Requesting Relief

Bolting is only accessible for examination when the CRDM is removed. Category B-0 requires that 10% of the peripheral CRDMs be removed each interval for examination.

In a subsequent response to a request for additional information, the licensee states the following in regard to the CRDM bolting:

The flange rings are actually a pair of semi-circular rings with threaded holes that comprise the "nut" portion of the bolted closure. Hence, they are in Category B-G-2 and included in the request for relief.

The bolts are approximately 17 feet below the deck of the cylindrical Head Service Structure. Seismic plates at the deck elevation obviate visual examination of the bolt heads from this location. The flange rings are positioned under the CRDM-to-head nozzle flanges and are not accessible.

There are eleven 12" diameter and one elliptical 12"x18" holes around the base of the service structure. A few of the outboard bolts on the peripheral closures could be examined after the cooling fans are removed from these 12" holes. We have greater confidence in relying upon the pressure test at 2255 psi that is performed after every opening of the Reactor Coolant System to substantiate the closure integrity, than visually examining a few of the 552 bolts every ten years.



It would be possible to view approximately 16 bolt heads via the remote T.V. on the adjacent closures to a removed CRDM. Again, we feel the pressure tests to be more effective.

After two CRDMs were removed for maintenance during the 1981 refueling outage, all 16 bolts were examined and there was no evidence of any service induced problems.

Evaluation

Due to the design of the reactor, the pressure retaining bolting and flange housing rings are located under the CRDM-to-head nozzle flange and are not accessible for visual inspection except when the CRDM is removed. Visual inspection of the bolting in place provides only limited information about the condition of the bolting. Furthermore, unbolting to examine the bolting may compromise the system more than it provides assurance of integrity. Evidence of leakage during hydrostatic pressure tests provides better information. The average total exposure encountered in each CRDM removal and reinstallation operation is 2000 mrem. The cost and personnel exposure encountered in removing all 69 of the CRDMs to make a visual inspection do not provide a compensating increase in safety.

The licensee does propose to examine the bolting whenever a CRDM is removed and is committed to remove 10% of the CRDMs each interval. This is an acceptable proposal if visual inspections for evidence of leakage are also made during pressure tests.

Conclusions and Recommendations

Based on the above evaluation, it is concluded that for the bolting discussed above, the Code requirements are impractical. It is further concluded that the alternative examination discussed above will provide necessary added assurance of structural reliability.

Therefore, it is recommended that relief should be granted from 100% visual examination of the CRDM bolting, provided that the bolting of all the removed CRDMs is examined and that visual examinations for evidence of leakage are made during hydrostatic pressure testing in accordance with IWB-5000.

References

Reference 9 (p 3, NDE-3); References 14 and 15.



4. Relief Request NDE-13, Integrally Welded Vessel Supports,
Category B-H, Item B1.12

Code Requirement

For vessel support skirts, the volumetric examination performed during each inspection interval shall cover at least 10% of the circumference of the weld to the vessel.

Code Relief Request

Relief is requested from making the volumetric examination.

Proposed Alternative Examination

None.

Licensee's Basis for Requesting Relief

The high radiation levels, 2 R/Hr outside the insulation surface, make performance of this examination contradictory to ALARA principles. The current edition of Section XI excludes this weld from examination, since it does not fall within the IWB boundary. See attached Figure IWB-2500-14 in the Reference 15 letter.

Evaluation

The 1977 Edition of Section XI through Winter 1979 Addenda has been referenced in 10 CFR 50.55a and inservice examinations may meet the requirements of this edition in lieu of those from previous editions with the following provisions:

- (a) Commission approval is required to update to the more recent edition (pursuant to 10 CFR 50.55a(g)(4)(iv)).
- (b) Any requirement of the more recent edition which is related to the one(s) under consideration must be met.

The Winter 1979 Addenda, Figure IWF-1300-1(c) shows a support weld outside the IWB, IWC or IWD boundary on a cast or forged attachment, as fitting into the IWF examination category. The requirement in Table IWF-2500-2 for this welded attachment is a visual VT-3 examination to determine the condition of the part, component or surface examined including such conditions as cracks, wear, corrosion, erosion or physical damage.



In a similar request for relief from the Oconee licensee, the NRC staff required that surface examination be performed on the inner surface of the weld as follows: (a) 10% of the circumference of the weld shall be examined, approximately sixty (60) inches, and (b) the areas examined shall consist of three twenty-inch lengths approximately 120° apart.

The visual examination of 100% of the weld is comparable in effectiveness to the surface examination of 10% of the weld so either method is acceptable. Since this weld is in compression, it is important that the inspection be done on the inside surface.

Conclusions and Recommendations

Based on the above evaluation, it is concluded that for the reactor vessel support skirt weld discussed above, the code requirements are impractical. It is further concluded that there is more than one alternative examination that will provide necessary added assurance and structural reliability. Therefore, it is recommended that:

- (1) Pursuant to 10 CFR 50.55a(g)(4)(iv), approval should be granted to update to the requirements of the Winter 1979 Addenda for Category IWF items. This approval would require a visual instead of volumetric examination of the weld.
- (2) Alternatively a surface examination may be performed as follows:
 - (a) 10% of the circumference of the weld shall be examined, approximately sixty (60) inches, and
 - (b) the areas examined shall consist of three twenty-inch lengths approximately 120° apart.
- (3) In either instance, the examination should be done on the inside surface.

References

Reference 15.

5. Relief Request NDE-4, Closure Head Cladding and Vessel Cladding, Category B-I-1, Items B1.13 and B1.14

Code Requirement

The examinations performed during each inspection interval shall cover 100% of the patch areas. The areas shall include at least six patches (each 36 sq. in.) evenly distributed in the closure head, and six patches (each 36 sq. in.) evenly distributed in accessible sections of vessel shell. The examination shall be 1) visual and surface or 2) volumetric for the closure head cladding, and visual for the vessel cladding.

Code Relief Request

Relief is requested from making the cladding examination.

Proposed Alternative Examination

None.

Licensee's Basis for Requesting Relief

Extremely high radiation levels are involved making performance of these examinations contradictory to ALARA principles. Requirements for performance of these inspections have been deleted from Section XI since Winter 1976. Radiation levels estimated from data obtained during plant outages and exposure levels are shown below:

<u>Component</u>	<u>Rad Level (R/hr)</u>	<u>Exam Time (hr)</u>	<u>Exposure (Man-Rem)</u>
Reactor Vessel Head	15	3	45
Reactor Vessel	15	6	90

Evaluation

The 1977 Edition of Section XI has been referenced in 10 CFR 50.55a and inservice examinations may meet the requirements of this edition in lieu of those from previous editions with the following provisions:

- (a) Commission approval is required to update to the more recent edition (pursuant to 10 CFR 50.55a(g)(4)(iv)).
- (b) When applying the 1977 Edition, all of the addenda through Summer 1978 Addenda must be used.
- (c) Any requirement of the more recent edition which is related to the one(s) under consideration must also be met.



The requirements for examining closure head cladding and vessel cladding are deleted from the 1977 Edition with Addenda through Summer 1978.

Conclusions and Recommendations

Based on the above evaluation, relief from Code requirements should not be granted. Instead, pursuant to 10 CFR 50.55a (g)(4)(iv), approval should be granted to update to the requirements of the 1977 Edition, Summer 1978 Addenda for Category B-I-1 items. This approval would delete the requirement to examine these items.

References

Reference 9 (pp 3 and 4, and NDE-4).

B. Pressurizer

1. Relief Request NDE-4, Vessel Cladding, Category B-I-2, Item B2.9

Code Requirement

The areas shall include at least one patch (36 sq. in.) near each manway in the primary side of the vessel. The visual examinations performed during each inspection interval shall cover 100% of the patch areas. These examinations may be performed at or near the end of the inspection interval.

Code Relief Request

Relief is requested from making the cladding examination.

Proposed Alternative Examination

None.

Licensee's Basis for Requesting Relief

Extremely high radiation levels are involved making performance of these examinations contradictory to ALARA principles. Requirements for performance of these inspections have been deleted from Section XI since winter 1976. Radiation levels estimated from data obtained during plant outages and exposure levels are shown below:

<u>Component</u>	<u>Rad Level (R/hr)</u>	<u>Exam Time (hr)</u>	<u>Exposure (Man Rem)</u>
Pressurizer	15	0.5	7.5

Evaluation

The 1977 Edition of Section XI has been referenced in 10 CFR 50.55a and inservice examinations may meet the requirements of this edition in lieu of those from previous editions with the following provisions:

- (a) Commission approval is required to update to the more recent edition (pursuant to 10 CFR 50.55a(g)(4)(iv)).
- (b) When applying the 1977 Edition, all of the addenda through Summer 1978 Addenda must be used.
- (c) Any requirement of the more recent edition which is related to the one(s) under consideration must also be met.



The requirements for examining vessel cladding are deleted from the 1977 Edition with Addenda through Summer 1978.

Conclusions and Recommendations

Based on the above evaluation, relief from Code requirements should not be granted. Instead, pursuant to 10 CFR 50.55a (g)(4)(iv), approval should be granted to update to the requirements of the 1977 Edition, Summer 1978 Addenda for Category B-I-2 items. This approval would delete the requirement to examine these items.

References

Reference 9 (pp 3 and 4, and NDE-4).



C. Heat Exchangers and Steam Generators

1. Relief Request NDE-4, Steam Generator Vessel Cladding,
Category B-I-2, Item B3.8

Code Requirement

The areas shall include at least one patch (36 sq. in.) near each manway in the primary side of the vessel. The visual examinations performed during each inspection interval shall cover 100% of the patch areas. These examinations may be performed at or near the end of the inspection interval.

Code Relief Request

Relief is requested from making the cladding examination.

Proposed Alternative Examination

None.

Licensee's Basis for Requesting Relief

Extremely high radiation levels are involved, making performance of these examinations contradictory to ALARA principles. Requirements for performance of these inspections have been deleted from Section XI since Winter 1976. Radiation levels estimated from data obtained during plant outages and exposure levels are shown below.

<u>Component</u>	<u>Rad Level (R/hr)</u>	<u>Examination Time (hr)</u>	<u>Exposure (Man-Rem)</u>
Steam Generator	15	1	15

Evaluation

The 1977 Edition of Section XI has been referenced in 10 CFR 50.55a and inservice examinations may meet the requirements of this edition in lieu of those from previous editions with the following provisions:

- (a) Commission approval is required to update to the more recent edition (pursuant to 10 CFR 50.55a(g)(4)(iv));
- (b) When applying the 1977 Edition, all the addenda through Summer 1978 Addenda must be used;
- (c) Any requirement of the more recent edition which is related to the one(s) under consideration must also be met.



The requirements for examining vessel cladding are deleted from the 1977 Edition with Addenda through Summer 1978.

Conclusions and Recommendations

Based on the above evaluation, relief from Code requirements should not be granted. Instead, pursuant to 10 CFR 50.55a (g)(4)(iv), approval should be granted to update to the requirements of the 1977 Edition, Summer 1978 Addenda, for Category B-I-2 items. This approval would delete the requirement to examine these items.

References

Reference 9 (pp 3 and 4, and NDE-4).



D. Piping Pressure Boundary

1. Relief Request NDE-5, Safe-End to Piping Welds, Category B-F, Item B4.1

Code Requirement

The volumetric and surface examinations during each inspection interval of 100% of each circumferential weld of dissimilar metals is required.

Code Relief Request

Relief is requested from the surface examination of the welds for the Core Flood A-Y Axis Nozzle and the Core Flood E-W Axis Nozzle.

Proposed Alternative Examination

Only a volumetric examination will be performed in accordance with frequency requirements of Category B-F with automated inspection equipment from the nozzle ID.

Licensee's Basis for Requesting Relief

Extremely high radiation levels involved make performance of a surface examination on Core Flood Nozzle Safe Ends contradictory to ALARA principles. Measurement of radiation levels in the areas of these welds has shown them to be approximately 10 R/hr. Examination time is estimated to be 3 hours. Therefore, total exposure for performance of these examinations is estimated to be 30 man-rem.

Evaluation

An exposure of 30 man-rem to conduct the surface examinations is not warranted by the increase in assurance of integrity of the weld. Environmental conditions tend to cause cracks to grow from the pipe inside diameter and along the heat affected zone of the weld. The licensee will be conducting a volumetric examination from the inside that will be effective in detecting any cracks in this area. The volume being examined could reasonably be expanded to include the area that would have been covered by the surface examination (covering the cross section bounded by ADFEDB, shown in sketch IWB-2500-8 of the 1977 Edition, Summer 1978 Addenda). This examination, together with the visual examination for leakage, will provide better information about the condition of the weld than the Code required examination.



Conclusions and Recommendations

Based on the above evaluation, it is concluded that for the welds discussed above, the code requirements are impractical. It is further concluded that the alternative examination discussed above will provide necessary added assurance of structural reliability.

Therefore, it is recommended that relief from the surface examination be granted, provided the expanded volumetric examination and visual examinations for leakage in accordance with IWB-5000 are performed.

References

Reference 9 (p 20, NDE-5).



2. Relief Request NDE-6, Support Attachments; Pressurizer Spray Decay Heat Removal Suction; Core Flood A; High Pressure Injection A, B, C and D; and Letdown Cooler; Category B-K-1, Item No. B4.9

Code Requirement

Volumetric examination during each inspection interval shall cover 25% of the integrally welded supports.

Code Relief Request

Relief is requested from the volumetric examination.

Proposed Alternative Examination

Surface examination.

Licensee's Basis for Requesting Relief

Volumetric examination of piping support attachment welds is impractical where the design does not employ a full penetration weld. In addition, later editions (beyond Winter 1976) permit either volumetric or surface examinations as applicable.

Evaluation

The 1977 Edition with Addenda through Summer 1978 permits either volumetric or surface examination as applicable, and thus the licensee could reasonably update to this Code version.

The 1977 Edition, Summer 1978 Addenda, of Section XI has been referenced in 10 CFR 50.55a and inservice examinations may meet the requirements of this edition in lieu of those from previous editions with the following provisions:

- (a) Commission approval is required to update to the more recent edition (pursuant to 10 CFR 50.55a(g)(4)(iv));
- (b) When applying the 1977 Edition, all the addenda through Summer 1978 Addenda must be used;
- (c) Any requirement of the more recent edition which is related to the one(s) under consideration must also be met.



Conclusions and Recommendations

Based on the above evaluation, relief from Code requirements should not be granted. Instead, pursuant to 10 CFR 50.55a(g)(4) (iv), approval should be granted to update to the requirements of the Summer 1978 Addenda of the 1977 Edition for Category B-K-1 items. This approval would permit either surface or volumetric examination as applicable.

References

Reference 9 (pp 34-36, and NDE-6).



E. Pump Pressure Boundary

1. Relief Request NDE-7, Reactor Coolant Pump, Pump Casing Welds, Category B-L-1, Item B5.6

Code Requirement

Volumetric examination shall cover the weld metal and the base metal for one wall thickness beyond the edge of the weld. The examinations performed during each inspection interval shall include 100% of the pressure-retaining welds in at least one pump in each group of pumps performing similar functions in system (e.g., recirculating coolant pumps). The examinations may be performed at or near the end of the inspection interval.

Code Relief Request

Relief is requested from volumetric examination of pump casing welds.

Proposed Alternative Examination

None until practical techniques are developed.

Licensee's Basis for Requesting Relief

No techniques are currently available for volumetric examination of thick stainless castings.

Evaluation

In their recent response to a request for information, Rancho Seco indicates that they are considering utilizing a device comparable to the MINAC System developed by EPRI and administered by J. A. Jones Applied Research Co.

The 1974 Code calls for an examination for a distance of one wall thickness on each side of the weld. With the MINAC or similar device this is impractical and examination of one-half inch on each side of the weld is adequate. This is in compliance with the 1977 Edition through Summer 1978 Addenda of Section XI.

The 1977 Edition of Section XI has been referenced in 10 CFR 50.55a and inservice examinations may meet the requirements of this edition in lieu of those from previous editions with the following provisions:



- (a) Commission approval is required to update to the more recent edition (10 CFR 50.55a (g)(4)(iv)).
- (b) When applying the 1977 Edition, all of the addenda through Summer 1978 Addenda must be used.
- (c) Any requirement of the more recent edition which is related to the one(s) under consideration must also be met.

Conclusions and Recommendations

Based on the above evaluation, relief from Code requirements should not be granted. Instead, pursuant to 10 CFR 50.55a(g)(4)(iv), approval should be granted to update to the requirements of the 1977 Edition, Summer 1978 Addenda for Category B-L-1 items. This approval would permit the examination of the weld metal and base metal for only 1/2 in. on each side of the weld. This approval would be required, assuming the licensee uses MINAC or a similar device for making a volumetric examination. If an examination by MINAC or a similar device is proved impractical, the licensee should submit another relief request.

References

Reference 9 (p 49 and NDE-7); References 14 and 15.

F. Valve Pressure Boundary
no relief requests.



II. CLASS 2 COMPONENTS

A. Pressure Vessels

1. Relief Request NDE-8, Decay Heat Removal Coolers, Nozzle-to-Vessel Welds, Category C-B, Item C1.2

Code Requirement

Volumetric examination of 100% of the nozzle-to-vessel attachment weld over the lifetime of the plant (four intervals and three periods within each interval) is required.

Code Relief Request

Relief is requested from performing the volumetric examination required by Code for the inlet and outlet nozzles.

Proposed Alternative Examination

Surface examination.

Licensee's Basis for Requesting Relief

The joint configuration of the inlet and outlet nozzle welds precludes a meaningful volumetric examination. The construction design utilizes a reinforcing ring plate as opposed to single full-penetration weld. Later code editions (beyond Winter 1976) require only surface examination for nozzles of this thickness.

Evaluation

The 1977 Edition of Section XI has been referenced in 10 CFR 50.55a and inservice examinations may meet the requirements of this edition in lieu of those from previous editions with the following provisions:

- (a) Commission approval is required to update to the more recent edition (pursuant to 10 CFR 50.55a(g)(4)(iv)).
- (b) When applying the 1977 edition, all of the addenda through Summer 1978 Addenda must be used.
- (c) Any requirement of the more recent edition which is related to the one(s) under consideration must also be met.

The 1977 Edition with addenda through Summer 1978 only requires a surface examination for nozzles in vessels 1/2 inch or less in nominal thickness.



Conclusions and Recommendations

Based on the above evaluation, relief from Code requirements should not be granted. Instead, pursuant to 10 CFR 50.55a(g)(v)(iv), approval should be granted to update to the requirements of the Summer 1978 Addenda of the 1977 Edition for Category C-B items. This approval would permit surface examination for nozzles 1/2 in. or less in nominal thickness.

References

Reference 9 (p 55 and NDE-8).



B. Piping

1. Relief Request NDE-9, Support Components, Category C-E-2, Item C2.6

Code Requirement

Visual inspection of the support components that extend from the piping, valve and pump attachment, including the attachment to the support structure.

Code Relief Request

Relief is requested from examining piping support components which are covered with insulation and do not show evidence of loss of support or inadequate restraint. Relief is not being requested for integrally welded supports (Category C-E-1, Item C2.5).

Proposed Alternative Examination

The insulation will be removed from a supported component for further inspections whenever an abnormality is detected that may have been a result of a loss of support capability or inadequate restraint.

Licensee's Basis for Requesting Relief

Permanent insulation will not be removed to perform the visual examinations of piping attachments, except in those cases where welded attachments are examined under Item C2.5 (Category C-E-1). It has been our experience that any loss of support capability or inadequate restraint can be detected through the inspection of the uninsulated portion of the support and the surrounding insulation. The governing Codes and Regulations used in the design and construction of the systems that are now Class 2 did not require provisions for inspection access. Thus, this requirement would place an undue additional inspection burden without a compensating increase in safety.

Evaluation

The mechanical connections to pipe straps are exposed and the licensee has committed to examine these connections. Possible damage to the portion of pipe strap covered by insulation can be detected by noting damage to insulation. The licensee has committed to remove the insulation and visually examine the pipe support in the event that inspection of insulation indicates pipe strap damage or loss of support. This is an adequate inspection.



Conclusions and Recommendations

Based on the above evaluation, it is concluded that, for the attachments discussed above, the Code requirements are impractical. It is further concluded that the alternative examination discussed above will provide necessary added assurance of structural reliability.

Therefore, it is recommended that relief be granted to eliminate visual inspection of those supports covered by insulation unless the inspection of insulation shows evidence of abnormality. This Code relief does not apply to integrally welded supports.

References

Reference 9 (p 78, NDE-9).

C. Pumps

No relief requests.

D. Valves

No relief requests.

III. CLASS 3 COMPONENTS

No relief requests.



IV. PRESSURE TESTS

A. General

No relief requests.

B. Class 1 System Pressure Tests

No relief requests.

C. Class 2 and Class 3 System Pressure Tests

1. Relief Request NDE-10, All Nuclear Class 2 and 3 Pump Suction and Discharge Piping Within Pump Isolation Valves (Class 2 and Class 3 Systems)

Code Requirement

The Code requirement for pressure testing Class 2 components is as follows:

- (a) The pressure retaining components shall be subjected to a hydrostatic test at 1.25 times the system design pressure at 100°F at least once toward the end of each inspection interval.

The Code requirement for pressure testing Class 3 components is as follows:

- (a) The system test pressure shall be at least 1.10 times the system design pressure.

Code Relief Request

Relief is requested from making the pressure test at Code required pressures.

Proposed Alternative Examination

Leak checks on piping during normal system operation on a schedule of once in ten years will be performed. (This would be the same type test as is performed on Nuclear Class 2 systems not normally in operation and on all Nuclear Class 3 systems every 40 months).

Licensee's Basis for Requesting Relief

Hydrostatically testing pump suction and discharge piping within pump isolation valves is not possible due to inability to protect pump seals from possible damage caused by the higher than normal hydrostatic test pressure.



Evaluation

The 1977 Edition of the Section XI has been referenced in 10 CFR 50.55a and inservice examinations may meet the requirements of this edition in lieu of those from previous editions with the following provisions:

- (a) Commission approval is required to update to the more recent edition (pursuant to 10 CFR 50.55a(g)(4)(iv));
- (b) When applying the 1977 Edition, all of the addenda through Summer 1978 Addenda must be used;
- (c) Any requirement of the more recent edition which is related to the one(s) under consideration must also be met.

The 1977 Code, Summer 1978 Addenda, allows the suction and discharge piping of a centrifugal pump, up to the first isolation valve, and the pump casing to be included in the pressure test for the suction lines. Updating to the newer code would also reduce the test pressure to a value below which seal damage should not occur and eliminate the need for relief from Code requirements.

Conclusions and Recommendations

Based on the above evaluation, relief from the Code requirements on hydrostatic pressure testing should not be granted. Instead, pursuant to 10 CFR 50.55a(g)(4)(iv), approval should be granted to update to the requirements of the 1977 Edition, Summer 1978 Addenda. This approval would reduce the pressure enough so that seal damage should not occur.

References

References 9 (p 5-10) and 12.



2. Relief Request NDE-11, Reactor Coolant Pump Seal Coolers and Piping (Class 3 Systems)

Code Requirement

The system test pressure shall be at least 1.10 times the system design pressure.

Code Relief Request

Relief is requested from making the examination at Code required pressures.

Proposed Alternative Examination

Leak check piping and coolers once in each inspection period.

Licensee's Basis for Requesting Relief

Hydrostatic testing of these coolers and piping would damage reactor coolant pump seals.

Evaluation

The 1977 Edition of Section XI has been referenced in 10 CFR 50.55a and inservice examinations may meet the requirements of this edition in lieu of those from previous editions with the following provisions:

- (a) Commission approval is required to update to the more recent edition (pursuant to 10 CFR 50.55a(g)(4)(iv));
- (b) When applying the 1977 Edition, all of the addenda through Summer 1978 Addenda must be used;
- (c) Any requirement of the more recent edition which is related to the one(s) under consideration must also be met.

Article IWD-5223(a) of the 1977 Code, Summer 1978 Addenda, provides for a piping section to be tested according to the setting of the lowest set relief valve that protects it. The pump seals should not be damaged using the criteria of the newer Code.

Conclusions and Recommendations

Based on the above evaluation, relief from the Code requirements on hydrostatic pressure testing should not be granted. Instead, pursuant to 10 CFR 50.55a(g)(4)(iv), approval should be granted to update to the requirements of the 1977 Edition, Summer 1978 Addenda. This approval would reduce the pressure enough so that seal damage should not occur.

References

References 9 (p 5-11) and 12.



3. Relief Request NDE-12, Auxiliary Feedwater Pump, Auxiliary Feed Pump Turbine Drive and Associated Mainsteam, Bearing Cooling and Seal Supply Piping (Class 3 Systems)

Code Requirement

The system test pressure shall be at least 1.10 times the system design pressure.

Code Relief Request

Relief is requested from making the examination at Code required pressures.

Proposed Alternative Examination

Leak check piping, pumps and turbine drive once in each inspection period.

Licensee's Basis for Requesting Relief

The Auxiliary Feedwater Pump Turbine Drive, its seals and the Auxiliary Feed Pump seals would be damaged by application of hydrostatic test pressure.

Evaluation

The 1977 Edition of the Section XI has been referenced in 10 CFR 50.55a and inservice examinations may meet the requirements of this edition in lieu of those from previous editions with the following provisions:

- (a) Commission approval is required to update to the more recent edition (pursuant to 10 CFR 50.55a(g)(4)(iv));
- (b) When applying the 1977 Edition, all of the addenda through Summer 1978 Addenda must be used;
- (c) Any requirement of the more recent edition which is related to the one(s) under consideration must also be met.

The 1977 Code, Summer 1978 Addenda, allows the suction and discharge piping of a centrifugal pump, up to the first isolation valve, and the pump casing to be included in the pressure test for the suction lines. Updating to the newer code would also reduce the test pressure to a value below which seal damage should not occur and eliminate the need for relief from Code requirements.



Conclusions and Recommendations

Based on the above evaluation, relief from the Code requirements on hydrostatic pressure testing should not be granted. Instead, pursuant to 10 CFR 50.55a(g)(4)(iv), approval should be granted to update to the requirements of the 1977 Edition, Summer 1978 Addenda. This approval should reduce the pressure enough so that seal damage should not occur.

References

References 9 (p 5-12) and 12.

V. GENERAL

No relief requests.



REFERENCES

1. R. W. Reid (NRC) to J. J. Mattimoe (SMUD), Rancho Seco, April 22, 1976.
2. J. J. Mattimoe (SMUD) to R. W. Reid (NRC), May 19, 1976.
3. R. W. Reid (NRC) to J. J. Mattimoe (SMUD), Rancho Seco Generating Station, November 17, 1976.
4. J. J. Mattimoe (SMUD) to R. W. Reid (NRC), January 24, 1978.
5. R. W. Reid (NRC) to J. J. Mattimoe (SMUD), February 15, 1978.
6. J. J. Mattimoe (SMUD) to R. W. Reid (NRC), February 27, 1978.
7. R. W. Reid (NRC) to J. J. Mattimoe (SMUD), May 30, 1978.
8. W. C. Walbridge (SMUD) to R. W. Reid (NRC), Proposed Amendment No. 63, Rancho Seco Nuclear Generating Station, Docket No. 50-312, March 16, 1979.
9. J. J. Mattimoe (SMUD) to R. W. Reid (NRC), Inservice Inspection and Testing Program, Rancho Seco Nuclear Generating Station, Unit No. 1, July 18, 1979.
10. Meeting between SMUD and NRC personnel, October 3 and 4, 1979.
11. R. W. Reid (NRC) to J. J. Mattimoe (SMUD), October 17, 1979.
12. J. J. Mattimoe (SMUD) to R. W. Reid (NRC), Rancho Seco Nuclear Generating Station, ISI/IST Program, December 10, 1979.
13. J. J. Mattimoe (SMUD) to R. W. Reid (NRC), Proposed Amendment No. 63, Revision 1, Rancho Seco Nuclear Generating Station, December 12, 1979.
14. J. F. Stolz (NRC) to J. J. Mattimoe (SMUD), Inservice Inspection (ISI), March 15, 1982.
15. J. J. Mattimoe (SMUD) to J. F. Stolz (NRC), Rancho Seco Nuclear Generating Station, Unit No. 1, April 19, 1982.



UNITED STATES NUCLEAR REGULATORY COMMISSIONDOCKET NO. 50-312SACRAMENTO MUNICIPAL UTILITY DISTRICTNOTICE OF GRANTING RELIEF FROM ASME CODE REQUIREMENTS

The U. S. Nuclear Regulatory Commission (the Commission) has granted relief from certain requirements of the ASME Code, Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components", to the Sacramento Municipal Utility District (the licensee), which revised the inservice inspection program for the Rancho Seco Nuclear Generating Station (the facility) located in Sacramento County, California. The ASME Code requirements are incorporated by reference into the Commission's rules and regulations in 10 CFR 50. The relief is effective as of the date of issuance.

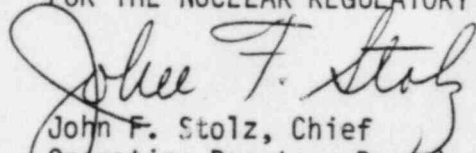
The requests for relief comply with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the letter granting relief and accompanying Safety Evaluation.

The Commission has determined that the granting of this relief will not result in any significant environmental impact and that pursuant to 10 CFR §51.5(d)(4) an environmental impact statement, or negative declaration and environmental impact appraisal need not be prepared in connection with this action.

For further details with respect to this action, see (1) the licensee's letters dated July 18, 1979, December 10, 1979, and April 19, 1982, (2) the Commission's letter to the licensee dated January 28, 1983, and (3) the Commission's related Safety Evaluation. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, N. W., Washington, D. C. and at the Business and Municipal Department, Sacramento City-County Library, 828 I Street, Sacramento, California. A copy of items (2) and (3) may be obtained upon request addressed to the U. S. Nuclear Regulatory Commission, Washington, D. C. 20555, Attention: Director, Division of Licensing.

Dated at Bethesda, Maryland, this 28th day of January 1983.

FOR THE NUCLEAR REGULATORY COMMISSION


John F. Stolz, Chief
Operating Reactors Branch #4
Division of Licensing