CERTIFIED BY:
Paul Shewmon - 4/5/93

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ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
JOINT SUBCOMMITTEE ON MATERIALS AND METALLURGY AND STRUCTURAL ENGINEERING
MEETING SUMMARY/MINUTES
MARCH 10, 1993
BETHESDA, MARYLAND

INTRODUCTION

The ACRS Joint Subcommittee on Materials and Metallurgy/Structural Engineering was held a meeting on March 10, 1993, in Room P-110, 7920 Norfolk Avenue, Bethesda, Maryland. The purpose of the meeting was to hear briefings on (1) steam generator tube degradation and tube interim plugging criteria (IPC), and (2) first-of-a-kind-engineering (FOAKE) piping design criteria for advanced light water reactor (ALWR) designs. The Designated Federal Official for this meeting was Elpidio Igne.

ATTENDEES: Principal meeting attendees included:

ACRS

Mr. P. Shewmon, Chairman
T. Kress, Member
C. Michelson, Member
I. Catton, Member
K. Seale, Member
J. Carroll, Member
P. Davis, Member
W. Lindblad, Member
J. Moulder, ACRS Consultant
E. Igne, Staff

NRC

H. Strosnider, NRR
C. Serpan, NRR
D. Terao, NRR

Others

R. Smith, Gas & Electric Corp.
D. Steininger, EPRI
M. Behravesh, EPRI
J. Santucci, EPRI
D. Rehn, Duke Power Co.
B. Moore, Southern Nuclear Operating Co.

DESIGNATED ORIGINAL

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MEETING HIGHLIGHTS, AGREEMENTS, AND REQUESTS

Opening Comments

Dr. Paul Shewmon, Chairman of the Joint ACRS Subcommittee on Materials and Metallurgy/Structural Engineering read the opening statement and provided a brief general background concerning the steam generator tube degradation concern. Dr. Shewmon stated that fine, short, outside diameter cracks within the tube support plate (TSP) have occurred in Inconel 600 steam generator tubes in some steam generators manufactured in the USA. The current staff tube plugging criteria states that all tube crack which penetrates more than 40 percent of the wall thickness needs to be plugged. This criteria has caused the licensees considerable maintenance problems because, for example, pinholes or small fine cracks in tubes exceeding 40 percent throughwall needs to be plugged or replaced. Many of these fine cracks are very difficult to detect using present eddy current (EC) tube inspection techniques. Many of these fine cracks, from the structural viewpoint, are not relevant in protecting the health and safety of the general public. Several plants have suggested an interim plugging criteria (IPC) for small fine outside diameter cracks for tubes within the tube support plate. The staff has accepted IPC based on eddy current coil voltage inspection techniques on a plant-by-plant basis. Dr. Shewmon stated that the Committee was briefed by the staff on the voltage-based criteria about a year ago.

STEAM GENERATOR TUBE DEGRADATION

EPRI Presentation

Steam Generator Degradation Specific Management (SGDSM), Mr. R. Smith, Senior VP, Rochester Gas and Electric Corporation on Behalf of EPRI's Steam Generator Reliability Project,

In his introductory remarks, Mr. Smith stated that the Ginna Nuclear Station steam generators will be replaced in the spring of 1996. He mentioned that the Ginna Nuclear Station steam generators tubes have not experienced fine short cracks at the support plate regions. In reply to Dr. Shewmon's question on the reasons why steam generator tube cracks are not experienced at Ginna Nuclear Station, Mr. Smith deferred the question to a later time.

Mr. Smith discussed a new concept initiated by industry in steam generator tube repair management that is under the auspices of the EPRI Steam Generator Reliability (SGR) project. Thirty domestic and seven foreign utility members belong to the EPRI SGR project. This project was formed because (1) conservatism of the current tube plugging criteria based on eddy current inspections, results in excessive tube repair/plugging, (2) degradation in the form of fine short cracks in steam generators containing mill-annealed Inconel 600 tubing at the drilled hole tube support plates is rapidly increasing, and (3) in many cases, such tube degradation does not compromise the structural integrity of the tube although the crack depth exceeds the current tube plugging criteria.

Mr. Smith stated that the degradation mechanism at the outside diameter of the steam generator tubes is caused by intergranular attack/stress corrosion cracking (IGA/SCC) mechanisms. Since about the early or mid-1980's the percentage of steam generator tubes plugged due to outside diameter degradation at the support plates in U.S. PWRs have been increasing rapidly. Dr. Catton mentioned that the Japanese is concerned with fluid elastic instability causing high cycle fatigue of the steam generator tube at the uppermost tube support plate. Dr. Catton stated that these loads must be accounted for in determining the integrity of steam generator tubes. The staff stated that fluid-elastic instability loads are accounted for and that some licensees have submitted documentation on this matter. These documents will be sent to us by the staff.

Mr. Smith stated that the program will allow some steam generator degraded tubes with acceptable safety margins to remain in service. He mentioned that the steam generator reliability program has already spent an excess of $3 million on developing elements of the Steam Generator Degradation Specific Management program and has committed substantial amount of manpower and funding resources, about $2-3 million, over the next three years.

Status of SGDSM Program, Mr. David Steininger, EPRI

Mr. Steininger, stated that on November 6, 1991, the ACRS Subcommittee on Materials and Metallurgy was briefed on the steam generator alternate tube repair criteria with representatives of the NRC staff and EPRI. An ACRS report dated November 15, 1991, was written stating that, "The continued use of the 40 percent depth limit as a repair limit results in a large effort by the licensees and a significant exposure to workers, and leads to the repair of many tubes that have a negligible risk of failure. We urge that the staff be encouraged to work with the industry to establish more appropriate and generic repair limits in a timely manner."

The staff, during the NRC Regulatory Information Conference, on July 21-22, 1992, indicated that, "In the staff's view, a matrix of flaw-specific measures could be developed that would consider flaw type, size, orientation and location. These flaw parameters would be analyzed to determine the appropriate corrective-measure strategy--which would integrate a specific repair (plugging or sleeving) criterion with a leakage rate limit and would specify augmented inspection requirements. The tube repair criterion would be adjusted to account for flaw growth rate and NDE uncertainty."

The industry has responded to the proposed NRC flaw mechanism matrix with a generic industry supported approach called SGDSM. This approach according to Mr. Steininger provides the generic requirements for degradation specific tube inspection, repair criteria development and implementation which can be used to produce the most cost-effective means to maintain acceptable plant safety. The SGDSM is a process for evaluating, monitoring and/or repairing tubes exhibiting distinct, individual tube degradation mechanisms.

Mr. Steininger discussed the structural and material design aspects of SGDSM. He mentioned that the margin against tube rupture involves the following elements: 1) tube rupture correlation, 2) structural limit and 3) repair limit. The tube rupture limit will use test data to determine the correlation between degraded tube rupture strength and the degradation specific inspection measurement parameter e.g., flaw depth, flaw length, flaw orientation, and voltage. Tube rupture data will be obtained from laboratory test of service degraded tube. The structural limit will be determined from the tube rupture correlation using the maximum pressure differential as the larger of 3.0 times the normal operation pressure or 1.4 times the postulated faulted load pressure. The repair limit will be defined as equal to the structural limit minus the inspection uncertainty and the tube flaw growth.

In reply to Committee questions, Mr. Steininger stated that outside diameter IGA/SCC occurs only at this specific location of the tube. For tube degradation at the free span, tube sheet or U-bend regions the current 40 percent through-wall plugging criteria still applies. Further, Mr. Steininger stated that the crack growth of the tube at the tube support plate tends to be limited because the tube support plate prevents tube radial growth in this region.

Mr. Steininger stated that with SGDSM, some small through-wall defects e.g., pinholes, can remain in service following inservice inspection. The interim repair criteria ensures that leakage during design basis accidents results in loses less than 10 CFR 100 limits. This requirement is met by demonstrating that the predicted accident leak rate from degraded tubes that remain in service is less than the plant allowable accident leak rate. Leak rate correlations will be obtained from inspected measured degradation and degraded tube leakage test results. It was mentioned by Mr. Michelson that loads due to a main steam line break accident should be accounted for in the degraded tube leakage correlation.

Mr. Steininger stated that a critical element of SGDSM involves in-service inspection (ISI) requirements. EPRI must implement an ISI guideline for steam generator examination based on random sampling strategy, augmented sampling, and examination based on its degradation mechanism and location. Further, Mr. Steininger stated that an inspection standard will be developed for the qualification of personnel, equipment and procedures for steam generator examination. Implementation of the SGDSM requires the utility to modify the plant Technical Specifications to incorporate degradation specific management to limit the allowable leak rate of 150 gallons per day per steam generator (presently at 500 gallons per minute) during normal plant operation.

The industry has submitted a generic SGDSM program to the NRC on December 1992. The staff recommended that the industry formally submit a lead-plant applicant. The industry will submit a plant-specific program by June 1993.

The following industry supported documents will require NRC review for SGDSM implementations:

- EPRI NP-6864-L Rev. 1, "PWR Steam Generator Tube Repair Limits: Technical Support Document for Expansion Zone PWSCC in Roll Transitions", December 1991

- EPRI TR-100407, "PWR Steam Generator Tube Repair Limits: Technical Support Document for Outside Diameter Stress Corrosion Cracking at Tube Support Plates", March 1992

- EPRI NP-6201, Rev. 3, "PWR Steam Generator Examination Guidelines", November 1992

These documents will be provided by EPRI for Committee use.

ISI Performance Demonstration Program, Dr. Mohammed Behravesh, EPRI

Dr. Behravesh, briefly discussed EPRI's steam generator inservice inspection performance demonstration program. In reply to a question concerning variability of the past NDE round robin examinations, Dr. Behravesh mentioned that EPRI has obtained preliminary results of a steam generator tube performance demonstration program involving 32 practicing ISI analysts (eddy current inspectors and interpreters). These analysts were asked to identify known steam generator tube degradation forms, such as, intergranular attack, pitting wear, primary stress corrosion cracking, and wall thinning. Preliminary results indicates that of the 32 analysts in the performance demonstration program, 20 of them have successfully passed. The results of the detection of intergranular stress corrosion cracking of degraded steam generator tubes correctly was quoted as 85 percent.

In reply to a question from Dr. Moulder, ACRS consultant, regarding flaw sizes, Dr. Behravesh stated that tube flaws in the demonstration program were of all sizes. Dr. Behravesh stated that the probability of false calls were about 4-5 percent, and that a 10 percent probability of false calls was cause for failure of the analyst during the demonstration program.

<u>Status Report of Farley Nuclear Plant Interim Plugging Criteria (IPC), Mr. B. Moore, Southern Nuclear Co.</u>

Mr. Moore, stated that the Farley Nuclear Plant was the first plant to submit IPC at the tube support plate region and specifically for the outer diameter stress corrosion cracking (ODSCC) degradation mechanism. He mentioned that this briefing updates the IPC briefing that was presented to the ACRS during the November 1991 meeting, and especially to mention the difficulty on Farley and other nuclear power plants in the industry are having by the staff not approving the IPC.

Mr. Moore mentioned that ODSCC of steam generator tubes at the support plates was detected during the 1985-1990 period during nuclear power plant inspections with the eddy current bobbin coil probe. Since 1987 Farley initiated a detailed steam generator management program, and 100 percent of the steam generator tubes were inspected, up from 3 percent of the tubes inspected at previous refueling outage. In reply to a question by Mr. Davis, Mr. Moore stated that bobbin coil probe inspection of all the tubes in a steam generator requires about 5 days, while inspection with a rotating pancake coil probe requires considerable more time. Mr. Moore mentioned that steam generator activities are generally on the critical path during refueling outages, and that since 1985, Farley has used two independent contractors in addition to there own inspectors to perform eddy current inspections of steam generator tubes.

Mr. Lindblad questioned Mr. Moore concerning whether the recent increase in steam generator cracks is due to better detecting inspection equipment or procedures or is the recent increase a process of tube degradation that we have had for many years but are only addressing now. Mr. Pitterle, of Farley, stated that tube degradation, especially due to ODSCC has been detected since the mid-1980's. These ODSCC growth have been tracked over the last three operating cycles and shows very small growth changes. The ODSCC was caused by unfavorable chemistry excursions that occurred during the mid-1980's. Farley Unit 1 became operational in 1977 and Unit 2 in 1981. Both plants operate with all volatile water chemistry. During the early years of operation, only a small sample of tubes were inspected and it was only during the mid-1980's did Farley institute 100 percent tube inspection with the bobbin coil probe and that tube degradation during the early plant operating years are difficult to trace. In the Trojan case, Mr. Pitterle stated that only two cycles of 100 percent inspection data exist and determining the origin of the cracks proved difficult.

Mr. Lindblad stated that based on his experience in the industry, there has been an intensified search for steam generator tube defects during the last several years as compared with just a routine search that may have been tolerated in earlier years. Further, he stated that the increasing number of defects we are presently experiencing in the field is due partly to the intensity of the search for cracks, as distinguished from tube degradation only recently occurring in steam generators. Mr. Pitterle mentioned that in 1987 a degraded steam generator tube was found to have a crack depth of 62 percent through-wall. Based on this experience the bobbin coil probe inspection guidelines were revised and is presently being used by EPRI. This change according to Mr. Pitterle, substantially increases steam generator tube crack detection.

Mr. Moore stated that aside from tube eddy current inspections, many steam generator tubes were removed from the steam generators. These tubes were analyzed and found to be degraded predominantly with ODSCC with minor amounts of IGA. Burst tests were also performed on these tubes. The lowest burst test pressure has been 5900 psi, which is well above the differential pressure of 4800 psi as required by Regulatory Guide 1.121. The burst pressure for a virgin tube was quoted to be in the range of 10,600 to 12,500 psi. In reply to a question by Mr. Michelson about tube loadings due to depressurization of a steam generator, Mr. Pitterle suggested that this topic should be deferred to another meeting.

Mr. Moore stated that they have submitted to the NRC an IPC to allow degraded tubes to continue to operate if the eddy current probe coil voltage is 4 (four) volts. Further, Mr. Moore believes that the staff will accept 3 (three) volts to be an adequate criteria when all the data are presented. This means that any steam generator tube inspected with a bobbin coil probe and found to be three volts or less at the tube support plate would be left in service. (Note: On a tube with no flaws or degradation the integrated voltage of the bobbin coil probe should read zero. Tubes with an increasing number of flaws or degradation should result in an increasing bobbin coil probe voltage). Mr. Moore stated that the Europeans have been using an IPC, in some cases, for about ten years. Both the Belgium and French have used IPC for steam generator tubes in areas of the tube sheet for primary water SCC degradation. They also allow through-wall and pinholes cracking in an free-span region of the tube if the allowable voltage is not exceeded. In the last three or four years both countries have been experiencing problems with ODSCC at the tube support plate (similar to Farley), and both have an IPC in place that allow on the order of 7-10 volt readings, (comparable American voltage reading) to stay in service. To date, they have had no problems with the IPC in terms of excessive leakage or tube burst concerns. Mr. Moore stated that they have over 100 reactor operating years of experience associated with their IPC.

In reply to a question by Mr. Carroll, Mr. Moore stated that a submittal by Farley on its IPC was docketed on February 1991. In the summer of 1992, the IPC was not yet approved by the staff and Farley was down for refueling outage. The staff has recently approved an IPC for one full cycle, for the Farley nuclear power plant that allowed degraded steam generator tubes to remain in service if the bobbin coil voltage is 1 (one) volt or less. Similar IPC's were also approved by the staff for Trojan, Catawba, and D. C. Cook. With the exception of Trojan, all plants are presently operating with the one volt IPC. Mr. Moore stated that Farley Unit 2 will need to have its approved IPC extended by the staff to support its fall 1993 refueling outage. In early 1993, Trojan nuclear power plant was permanently shutdown, attributed in part to issues relating to not having an IPC in place and the uncertainty of obtaining an approved IPC. Mr. Moore stated that by not having the IPC in place, the industry is having to unnecessarily plug or sleeve degraded steam generator tubes, the industry believes to be adequate. Both of these steam generator maintenance operations are man-rem extensive and costly. It was mentioned that when sleeving degraded tubes, tube inspection becomes more of a problem. Mr. Moore mentioned that with the IPC in place, the industry should be able to satisfactorily manage the tube degradation problems, and allow the operation of steam generators to their full useful life and not unnecessarily have to replace the steam generators at a cost of $150-200 M per unit.

In reply to a question concerning secondary side water chemistry, Mr. Hundley, of Farley, stated that in the past, the philosophy of using pure water was prudent. It was mentioned by Dr. Shewmon that Alloy 600 material in pure water would not eliminate stress corrosion cracking, according to Dr. Correo.

In reply to a question by Dr. Shewmon concerning the basis of the proposed 3 volts criteria, Mr. Moore stated that the Farley IPC of 3 volts is based on the correlation of the bobbin coil voltage vs. burst pressure for degraded steam generator tubes removed from service and corrected for Farley specific inservice inspection uncertainties and crack growth rate. It was mentioned that the staff favors a one volt IPC instead of three volts.

NRR Presentation

Mr. J. Strosnider, presented the staff's perspectives on issues regarding degradation of steam generator tubes. Mr. Strosnider mentioned that this issue continues to be a major problem that results in forced outage, extends planned outages, add personnel radiation exposure and is costly. Ten nuclear power plants have replaced steam generators and North Anna Nuclear Power Plant is now replacing theirs.

The present technical specification on when steam generator tube needs to be plugged is defined as when defects, such as, cracks, wastage, pittings is equal to or exceeds 40 percent of the tube wall thickness. Presently, licensees are typically inspecting more tubes than required by the technical specification and that licensees are using enhanced inspection methods. There are no requirements on the use of multiple frequency or rotating pancake coil probes to perform the tube inspections. Mr. Strosnider stated that based on what is occurring in the field, the NRC requirements need to be updated to reflect operating experience, and advances in tube inspection technology. For example, the industry is focussing on a tube plugging criteria based on eddy current coil voltage. Mr. Strosnider emphasized that the voltage-based plugging criteria is intended to address only IGA and axial ODSCC of the steam generator tube at the tube support plate intersections. The industry has proposed an IPC based on the voltage measured across a bobbin probe coil of 3 volts. He mentioned that the staff to date has approved an interim plugging criteria for one fuel cycle of operation, to a more restrictive one volt limit. The staff has also approved enhanced inspection methods to minimize the testing variability. In response to a Subcommittee question concerning the 3 volts vs. 1 volt criteria, Mr. Strosnider stated that the staff has concluded that the 1 volt criteria provided adequate margin, while the 3 volt criteria does not provide, at this time, for adequate margins.

Mr. Strosnider stated that the staff has also approved lower technical specification primary-to-secondary leakage rate limits. This lower limit is in recognition of the fact that through-wall cracks may exist in tubes left in service when using the voltage-based criteria. Mr. Strosnider pointed out that Trojan, which was recently shut down, had a tube leak greater than its technical specification limit which caused the operators to shut down the plant. It was mentioned that Trojan was in operation based on an interim plugging criteria of 1 volt and was shut down because of excess leakage at a tube sleeve joint. The tube sleeve joint leaked because of inadequate annealing.

Mr. Strosnider stated that in February 1993 the Interim Plugging Criteria (IPC) task group was formed to review the technical bases related to voltage based IPC for ODSCC at the tube support plate intersections. A summary of the task group findings will be documented and sent to the ACRS when the work is complete.

Mr. Strosnider next discussed the postulated main-steam-line-break (MSLB) leak rate determination. The major areas of concern are reactor water storage tank (RWST) inventory and off-site doses. The industry model predicts negligible MSLB leakage while the NRC model predicts leakages ranging from 33 gpm to 1350 gpm (best estimates of about 145 gpm) at a pressure differential of 2600 psi. He mentioned that the NRC model is overconservative compared with operating experience. The IPC task group is investigating the differences.

Mr. Strosnider discussed the staff's future plans in this area. A regulatory framework for implementing defect specific management of steam generator tube degradation will be developed. The staff prefers a generic approach by the industry in order to provide consistency among licensees and to minimize the impact of staff resources. Further, additional confirmatory research on steam generator tube integrity issues are needed.

RES Presentation, Mr. C. Serpan

Mr. Serpan, in his opening statement, stated that NRR and RES have been interacting on the degraded steam generator tube issue for a long time. Mr. Serpan outlined a proposed research program to provide technical basis for tube crack growth rate, NDE detection reliability and sizing accuracy, and the correlation of voltage criteria with leak rate and burst pressure.

Mr. Serpan mentioned the research program being done by Oak Ridge National Laboratory (ORNL) for the past several years. The objective of the program is to enhance regulatory acceptance of inspection by advanced eddy current test methods and equipment. Eddy current probes are being developed to account for: 1) tube diameter changes due to denting, 2) the presence of tube support plates during tube inspection, 3) changing wall thickness of tubes, and 4) differences between IGA or IGSCC tube degradation. A research program for steam generator tube inservice inspection is being performed at the Pacific Northwest Laboratory (PNL). The objective of the PNL research program is to develop an independent performance demonstration criteria for eddy current inservice inspection for the qualification of personnel, procedures and equipment. Mock-up of degraded steam generator tubes are being built for use by Region I in order to evaluate the quality of inservice inspection being performed. Mr. Strosnider noted that Regulatory Guide 1.83, "Requirements for Test Samples, Inspector Examinations and Grading Criteria," will be updated.

Mr. Serpan described the future NRC steam generator tube integrity research program. Some elements of this planned program are 1) perform leak rate tests of tubes with various cracks under normal and MSLB conditions, 2) validate models to predict leak rate of degraded tubes, 3) confirm eddy current voltage response from realistic cracks, 4) validate models to predict failure modes of degraded tubes, and 5) integrate results in order to evaluate IPC proposals and to prepare revised regulatory guides, technical specifications and regulations. In reply to a question by Mr. Carroll on when the results of this program will be available to evaluate the IPC proposals, Mr. Serpan stated that the results of the future or long-term research program will not be available for several years. But stated that a very strong staff effort under way now will provide good answers for the interim plugging criteria. Further, Mr. Serpan stated that as research results become available the criteria will be refined and upgraded.

In reply to a question by Mr. Lindblad regarding the interim plugging criteria of one volt, Dr. B. Liaw, NRR, replied that depending of the research program results the one volt criteria could increase to two, three or four volts rather than to have it decreased to below one volt.

Mr. Carroll commented that the licensees ought to know the condition of their steam generators and have a strong incentive to do the proper surveillance/maintenance in order to maximize reliability and reduce down time and cost.

FIRST-OF-A-KIND-ENGINEERING (FOAKE) FOR SEISMIC PIPING DESIGN CRITERIA

Industry Presentation

Introductory Comments, Mr. J. Santucci, EPRI

Mr. Santucci, stated that the Advanced Reactor Corporation (ARC) is a non-profit organization, representing utility interests in the development of advanced light water reactors (ALWR). Under the auspices of the Nuclear Power Oversight Committee, ARC primary responsibility is the design of standard commercial building blocks for advanced nuclear power plants. Mr. Santucci mentioned that two plants, the AP600 by Westinghouse and the ABWR by General Electric, have been selected for detail building blocks studies for the next three years at a cost in excess of $200 million. The goal is to provide a level of detail design completion for these plants to about 60-70 percent, or substantially beyond that required for design certification. One of the generic issues that they felt could provide a substantial improvement in the design process of the first-of-a-kind engineering (FOAKE) for ALWR is seismic piping system design criteria which will be discussed today.

Background Information, Mr. D. Rehn, Duke Power

Mr. Rehn stated that the ALWR piping systems should be reliable, safe, and meet the NRC regulations and ASME codes. Mr. Rehn further stated that the ASME code on piping design is out of date, over conservative, and do not reflect current technical understanding of piping systems behavior. These factors have driven the nuclear industry to produce unnecessarily conservative and costly piping systems designs. The FOAKE seismic piping system design criteria project objective is to facilitate the establishment of appropriate changes in the ASME Code criteria and design practice in order to improve safety, reliability, and to reduce cost and construction time for ALWR piping and support systems. Mr. Rehn discussed work done in the early 1980's by the Pressure Vessel Research Committee and the Steering Committee on Piping System on damping values, spectral broadening, piping installation tolerances and dynamic stress allowables. In reviewing past work on piping design, Mr. Rehn, mentioned that insufficient experimental, analytical and test data were available to reach satisfactory conclusions. In the mid-1992 the Technical Core Group was formed by ARC to optimize piping system design and to prepare design specifications. This group was headed by Mr. D. Landers, Teledyne. The Technical Core Group has initially gathered test data and developed analytical techniques in order to provide the background necessary to propose a series of code revisions and design guidelines. In response to a question by Dr. Shewmon, Mr. Rehn stated that the proposed design guidelines will utilize the existence of the material higher damping factor when operating in the elastic-plastic range. In general, Mr. Rehn stated that the design guidelines will provide a less rigid piping system than present piping system, designed using present codes and standards.

Design Margins, Mr. S. Tagart, EPRI

Mr. Tagart described the Piping and Fitting Dynamic test program that was conducted from 1985-1988 and was managed by EPRI and the NRC. The program was designed to present ASME codes and standards. Three types of tests were performed. They are component tests, piping systems tests, and materials tests. The component tests were run to determine the component behavior and failure modes. The system tests confirmed redistribution loads, failure modes, and provided insight to the propose code changes. The materials tests were performed to study the failure modes of different piping materials under large dynamic loads.

The major conclusions from the tests are listed as follows:

- Dynamic load reversals prevents pipe collapse
- Excessive pipe ratcheting occurs when the dynamic loads are much greater then the SSE loads
- For large dynamic loads the piping system damping factors are much larger than those found in Regulatory Guide 1.61
- Elastic analysis using 5% damping and ± 15% broadening is conservative at the proposed Level D stress limits
- Seismic margins were established from the seismic tests
- Minimum margin for piping system designed using the present ASME code is about 10.6
- Piping failure mode is by fatigue/ratcheting and not by pipe collapse as assumed in the ASME code
- Internal pipe pressure prevents pipe collapse during seismic occurrence

Mr. Tagart stated that the Te hnical Core Group has proposed ASME code changes to reflect the results of the seismic margin tests. The proposed piping design guideline was submitted to the ASME for consideration in 1988. After about two years of evaluation and deliberation the ASME Working Group on Piping on a straw vote, voted in favor of the proposed guideline by a 5-1 margin; the negative vote was from the NRC. Mr. Tagart mentioned that a consulting company is now reviewing the test program data for NRC, and will meet with them later this month.

Examples from the Design Margin Study, Mr. E. Brands, Sergeant & Lundy

Mr. Branch discussed piping margins obtained during the test program. He stated that using the current ASME code (1989 amend.) for Class 1 piping system and assuming a 2 percent damping factor, the seismic margin is 10.6. Using the proposed Technical Core Group proposed design criteria, which account for an increase damping factor, provides an increase in allowable stress, and using a frequency effect reduction factor, result in a new calculated seismic margin of 3.3.

<u>New Piping Design Criteria,</u> Mr. D. Landers, Teledyne

Mr. Landers presented proposed new piping design criteria for piping system design. He stated that water/steam hammer loads are not accounted for in the proposed piping design criteria. Only seismic or reverse dynamic loads superimposed on pressure loads are applicable for the proposed new piping design criteria. As long as the dynamic loads are reversing, these proposed criteria are applicable.

Presently, the ASME code protects against collapse or plastic instability. The Technical Core Group is proposing that the ASME code should reflect the failure mode witnessed during the tests, namely fatigue or ratcheting enhanced fatigue. In reply to a question by Mr. Lindblad on the protection of piping system subject to water or steam hammer loads, Mr. Landers stated that the ASME code already have adequate design provisions to account for these loads. In most cases, Mr. Landers stated that the piping designer do not account for these dynamic hydraulic loads during the design phase, which caused a few piping system to experience failures. Mr. Liaw, NRC, stated that the issue of accounting for the unexpected water hammer loads, for example, is why the NRC staff is in dispute with the proposed seismic pipe design criteria. Mr. Landers stated that piping system design should not use seismic rules to design for water/steam hammer loads.

Mr. Landers mentioned that the ALWR plants will be designed only for the safe shutdown earthquake (SSE) loading event, as the operating basis earthquake (OBE) loading event will be eliminated as a design basis. Mr. Landers stated that, in his opinion, this elimination of the OBE as a design basis event is a good one.

The proposed rules for SSE design by analysis, requires the use of inelastic dynamic analysis. To protect against the failure mode experienced during the test program, the average strain through the wall thickness will be limited to protect against the ratcheting swelling effects, and that the peak single amplitude strain will be limited to protect against fatigue failure. In performing this analysis all loading occurring at the same time as the earthquake must be considered e.g., pressure, thermal expansion, dead weight, water hammer, etc. In actuality, design by analysis today is very difficult and the use of margin methodology or simplified rules should be used instead. The proposed simplified rule process is as follows:

- Requires linear-elastic response spectrum analysis with ±15% spectrum broadening and using a 5% material damping value

- Limits the average stress through the wall due to internal pressure

- Limits the bending stress due to weight effects
- Limits the stress due to pressure, dead weight and inertial earthquake loading
- Requires an increase of calculated support loads
- Requires that calculated displacements meet Design Specification Limitations

Summary/Conclusions, Mr. D. Rehn, Duke Power

Mr. Rehn stated that we now have a better understanding of the behavior and failure mode of seismic piping systems than we did 10 years ago. This knowledge must be used to modify the present codes and regulations in order to design a more reliable and economic piping system. The Technical Core Group has proposed a revision to the ASME Section III Code.

NRC Presentation on ALWR Piping Design Criteria, Mr. D. Tarao, NRR

Mr. Tarao discussed past regulatory concerns and piping issues. Some of the issues are 1)OBE controlled piping design, 2) excessive number of seismic restraints, and 3) a large number of pipe rupture mitigation devices. Mr. Tarao stated that based on extensive NRC activities affecting piping system design e.g., NRC Piping Review Committee (1983), Piping Dynamic Tests, earthquake experience data, etc., the NRC have instituted many regulatory activities, such as, approval of ASME Code Cases N-411, "Damping Values," and N-397, "Peak Shifting," use of independent support motion method (NUREG-1061), higher piping functional capability limits (NUREG-1367), and elimination of OBE from design (10 CFR Part 100).

Mr. Tarao stated that the staff is evaluating the safety margins in piping system design. The staff has not been requested to review the ARC (FOAKE) piping criteria, but Mr. Tarao mentioned that the NRC will review the ASME code changes and will endorse the ARC piping criteria through rulemaking (10 CFR 50.55a), if found adequate.

In a reply to a question by Mr. Michelson, Mr. Tarao stated that the leak-before-break approach has not been applied to piping system outside the containment e.g., main steam lines, although Mr. B.D. Liaw stated that leak-before-break approach can be applied to piping system outside the containment.

Dr. Shewmon thanked the presenters and mentioned that the subcommittee looks forward to the day when the piping design criteria is finalized and presented for our review.

Dr. Shewmon noted that the presenter for Pacific Gas & Electric Corp. could not be present for today's meeting, but that the PG&E presentation material was submitted for the record.

The meeting was adjourned at 4:45 p.m.

FUTURE SUBCOMMITTEE ACTIONS AND ITEMS FOR FOLLOW-UP

Future Subcommittee Actions

This meeting was a briefing only and the Subcommittee decided not to brief the full ACRS Committee during the April 1993 meeting. The Subcommittee would like the opportunity to review the staff's steam generator interim plugging criteria after the public comments have been reconciled and before its release in final form. Further, the Subcommittee would like to be periodically updated on the status of the revisions to the ASME Section III Code.

Follow-Up Items

1. Mr. Smith, Rochester Gas & Electric Corp., deferred a Subcommittee question on why Ginna's steam generator tubes are not cracking.

2. The staff will transmit to us, licensee submittals on fluid-elastic instability loads that cause high-frequency cyclic fatigue damage/failures to steam generator tubes.

3. The Subcommittee requested the following documents for information:

   - EPRI NP-6864-L Rev. 1, "PWR Steam Generator Tube Repair Limits: Technical Support Document for Expansion Zone PWSCC in Roll Transitions," December 1991

   - EPRI TR-100407, "PWR Steam Generator Tube Repair Limits: Technical Support Document for Outside Diameter Stress Corrosion Cracking at Tube Support Plates," March 1992

   - EPRI NP-6201, Rev. 3, "PWR Stream Generator Examination Guidelines," November 1992

4. Mr. Strosnider, NRR, will send to the Subcommittee, the document on the findings of the Interim Plugging Criteria Task Group when completed.

BACKGROUND MATERIALS PROVIDED FOR THIS MEETING

1. Regulatory Guide 1.83, "Inservice Inspection of Pressurized Water Reactor Steam Generator Tubes," July 1975, Rev. 1

2. Regulatory Guide 1.121, "Bases for Plugging Degraded PWR steam Generator Tubes," August 1976

3. SECY-92-412, Trojan Steam Generator Issues," December 16, 1992

4. Memorandum for Thomas E. Murley, NRR, from Eric S. Beckjord, RES, Subject: "Interim Plugging Criteria for Trojan Nuclear Plant," January 5, 1993

5. Memorandum for Thomas E. Murley, NRR, from Eric S. Beckjord, RES, Subject: "Interim Plugging Criteria for Trojan Nuclear Plant," January 15, 1993

6. Note to T. Murley, E. Beckjord and J. Martin, NRC, from Steve Hanover, Subject: Report on Trojan Restart, January 19, 1993

7. Information Notice 92-08: Operation with Steam Generator Tubes Seriously Degraded, December 7, 1992

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NOTE: Additional meeting details can be obtained from a transcript of this meeting available in the NRC Public Document Room, 2120 L Street, NW, Washington, DC 20006, (202) 634-3273, or can be purchased from Ann Riley and Associates, Ltd., 1612 K Street, NW, Suite 300, Washington, DC 20006, (202) 293-3950.