ES-LPD-82-007

CRBRP TRANSITION JOINTS

CRBRP ENGINEERING STUDY REPORT

APRIL 1982

WESTINGHOUSE ELECTRIC CORPORATION
ADVANCED REACTORS DIVISION
P.O. BOX 158
MADISON, PENNSYLVANIA 15663

Compiled by:

S. Diamond

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W-ARD

Approved by:

R H Mallett

R. H. Mallett, Manager
CRBRP Piping Design and
Mechanical Equipment

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ABSTRACT

This report provides responses to specific NRC questions on CRBRP transition joints. Industrial and nuclear experience with transition joints is reviewed, the technical basis for the CRBRP applications of the joints is provided, the fabrication, environment, and service conditions of the joints are described, and the methodology and results of the analyses of the joints are presented.

It is concluded that prior experience with transition joints provides sufficient understanding to fabricate and evaluate the CRBRP transition joints and also reinforces the confidence that the joints can meet their service requirements.

It is concluded that the design bases for the reactor vessel transition joint were conservative, the environment to which it is exposed is benign, and the service conditions will be met by the present design.

TABLE OF CONTENTS

1.0 SUMMARY

Information is presented on transition joint technology and experience, on the design, analysis, environment and service conditions of the CRBRP reactor vessel transition joint, and on the IHTS transition joint design evaluations in response to NRC questions CS 210.3 and CS 250.5.

It is concluded that a large body of prior experience with transition joints provides sufficient understanding to proceed with the design, fabrication and evaluation of the CRBRP transition joints.

It is further concluded that the design, evaluation, and imposed service conditions yield an appropriate reactor vessel transition joint of high integrity.

Finally it is shown that the design and analysis of the IHTS transition joints are sufficiently comprehensive to assure that the integrity of the joints will be maintained.

2.0 INTRODUCTION

A series of questions was sent to the CRBRP Project to address concerns about intended CRBRP materials, high and low temperature regions of the plant, design and analyses approaches, and specific welded joints in the plant, i.e., the reactor vessel transition joint and the IHTS transition joints.

There were two specific questions concerning transition joints, CS 210.3 and CS 250.5, which are presented below:

CS 210.3 Thermal expansion and creep rate generally vary among different materials. Describe methods and procedures used to evaluate local stresses and strains at places where Bi-metallic and Tri-metallic transition welds are applied.

CS 250.5 Provide the method and data base for the structural evaluation and acceptance of Bi-metallic and Tri-metallic transition welds for service in the primary and intermediate heat transport systems.

To address these and other questions a CRBRP/NRC meeting was held at Bethesda, Maryland on April 6-7, 1982 at which there were three topical discussions concerning transition joints. Figures 1, 2 and 3 were used in introducing these discussions.

This report provides the responses to the above NRC questions.

CRBRP HTS MATERIALS AND STRUCTURES

IX. Transition Joint Experience★

- Purpose
  - To establish awareness of service experience with transition joints
- Scope
  - Joint designs
  - Service conditions
  - Performance
- Conclusion
  - Prior experience provides sufficient understanding to enable creation/evaluation of the CRBRP transition joints

★ Responds to Q CS 210.3, CS 250.5

FIGURE 2.0-1

CRBRP HTS MATERIALS AND STRUCTURES

X. Reactor Vessel Transition Joint★

- Purpose
  - To describe the design of the low temperature reactor vessel transition joint
- Scope
  - Geometry, material and fabrication
  - Conditions and loadings
  - Structural evaluation
- Conclusion
  - The engineered design and the service conditions establish integrity of the RV transition joint

★ Responds to Q CS 210.3, CS 250.5
Related to PSAR 5.2.6

FIGURE 2.0-2

7010-13

CRBRP HTS MATERIALS AND STRUCTURES

XI. IHTS Transition Joints★

- Purpose
  - To describe the design of the IHTS transition joints
- Scope
  - Geometry, material and fabrication
  - Conditions and loadings
  - Structural evaluation
  - Verification testing
- Conclusion
  - The design program is sufficiently comprehensive to ensure integrity of the elevated temperature IHTS transition joints

★ Responds to Q CS 210.3, CS 250.5

FIGURE 2.0-3

TRANSITION JOINT EXPERIENCE

Presented by
P. Patriarca (ORNL)
and
G. M. Goodwin (ORNL)

TRANSITION JOINT EXPERIENCE

This first discussion (Figure 1) summarizes the extensive background of transition joint technology as a basis for the discussion of the specific design of the CRBRP transition joints.

The CRBRP heat transport system (Figure 2) includes the pressure vessel and transition joint and a steam generator transition joint circuit.

I will first discuss the reactor vessel system and then the steam generator (Figure 3). The experience that I will review is that which is applicable to either end of the circuit, from a fossil energy standpoint, from a light water reactor standpoint, the LMFBR standpoint as we now know it today, using some foreign experience, talking about joint designs and performance.

It will be concluded, as shown in Figure 3, that prior experience provides sufficient understanding to evaluate the CRBRP transition joints.

LWR EXPERIENCE

In Figure 4 we have the definition of the first problem, which is our reactor vessel transition joint. It is a transition between A508 steel through an intermediary alloy, Inconel 600, and ultimately to 304 stainless steel, the connection being made with an Inconel type weld metal called 82, which is a bare wire and can be used either with gas tungsten arc or with submerged arc. B&W used the submerged arc process. The numbers describe the coefficient of expansion at 450°F, which is the nominal ambient operating temperature of the pressure vessel.

Figure 5 shows the relevant experience in the light water industry that operates in the saturated steam cycle. This is a typical nozzle with the 508 nozzle connected to a safe end.

The safe end, normally speaking, is 304 SS, could be Inconel 600, and in some cases, (Combustion Engineering's preference) stainless clad steel. The normal procedure is to butter the interface on the A508 vessel with an Inconel 82, bare wire with GTA, sometimes with stick electrodes like 182, then heat treat the vessel with the safe end attached. Current practice is to machine the butter and attach the safe end in place, after the vessel has been heat treated, with Inconel weld filler metal.

There have been times when fabricators welded the 304 safe end onto the vessel and heat treated the vessel. This resulted in sensitization of the stainless steel, which in some applications is considered objectionable. The normal practice now is the former.

Then, of course, the safe end is connected in the field by the fabricator to whatever stainless steel circuitry he may have with conventional stainless steel welding.

This joint is used in a pressurized water reactor between the reactor vessel and the piping and at the other end from the piping to the steam generator, and in the boiling water reactor at the exit from the primary vessel. So far as we know, these joints have been operated to the satisfaction of just about everybody. There have been no reported failures.

Figure 6 presents experience for pressurized water reactors that are about ten years old or older, assuming that they have been on line between 50 and 80 percent of the time and that the nozzles have been subjected to load and temperature for that increment -- and of course they vary in time from about eleven years on line, nine years on line, to twenty-five years on line and they have operated very satisfactorily.

As pointed out earlier, the other end of the circuit shows that the PWR steam generator nozzles behave in a comparable fashion without any difficulty.

The boiling water reactor experience is shown in slide 7. Of the twelve shown two are demonstration plants, and the rest of them are producers. They vary from nine to twenty-two years of service without any reported failures. So there are approximately twenty large diameter transition joints of the variety that are comparable to the Clinch River transition pressure vessel joint, which have had a lengthy service to the satisfaction of everyone.

Figure 8 illustrates that the choice of Inconel was predominantly for coefficient of thermal expansion transition. There is an added advantage in that, if you use an Inconel butter, when you heat treat the A508 vessel at 1100 plus degrees Fahrenheit for a while, being a nickel-based material, the diffusion of carbon into the nickel-based material would be minimal.

But the important thing is that the transition from the stainless steel in the case of Clinch River is through an Inconel 82 weld metal, which would have a coefficient of expansion of about 7.8, to an Inconel 600 spool, which would have a coefficient of expansion at these temperatures of about 7.7, once again to Inconel 82, to the A508, which has an expansion of 6.9. So you would assume that the interface, where everything usually happens, has material with an expansion coefficient of 6.9 versus 7.8, which is a close match. It is comforting to know that you made that attachment in that fashion, and that there should be no problems associated with it.

That ends my discussion on the light water reactor service, applicable to the CRBR reactor vessel joint.

LMFBR EXPERIENCE

Figure 9 defines the CRBRP intermediate heat transport system. On the right-hand side, up at the superheater, joint number 1, which is the inlet to the superheater, 26-inch diameter pipe about a half inch wall, -- steam exits at about 936° and sodium at a little higher than that.

The evaporator outlet, which is below, is about an 18-inch diameter piece of pipe. Its temperature is considerably lower, in the range of 650°F. This is one of the large-diameter pipes. So when we address that problem, we ask the question, what do we know about the service conditions for it?

Most of the LMFBR's that have been running around the world today, like the PFR in the U. K. since 1977, have a stainless steel superheater. So there are no transition joints as such in those reactors.

In the other circuits, in the evaporators and so on, they have transition joints but they are at a lower temperature. Phoenix, which has been running since '74, also has a stainless steel superheater. BN-600 also uses a stainless steel superheater.

The projected Super Phoenix, which is targeted for 1984, employs an alloy 800 superheater. So it won't have a transition joint as such.

For Super Phoenix II (SNR II), there is an option of using for the superheater EM-12, which is a 9 Cr-2 Mo alloy. In that case it would have a transition joint. So the French metallurgists are thinking about the ultimate use of a transition joint.

The stainless steel superheaters in PFR cracked extensively and are going to be replaced with 9 Cr-1 Mo. And their commercial thoughts are also to go 9 Cr-1 Mo.

So they are thinking overseas of transition joint situations, but how they are going to approach them, we don't know yet.

Figure 10 shows the applicable steel superheaters like Fermi which lasted ten years. The EBR-II is still on line 19 years later. And the BN-350 on the Caspian Sea, partially desalting and partially power, has been on line for about 9 years. SNR, which is going to be on line shortly, has 2-1/4 chrome one moly and stabilized superheaters and therefore has a transition joint. Service temperatures are 820, 815 and 780°F. These are not as high as in CRBRP (936°F), but they are high enough for comparison. Looking at their experience, none of them have reported any problems.

We look specifically at the EBR-II, our own home-grown version of a successful breeder reactor, in Figure 11. This joint design is attributed to Bob Nolan and Cecil Stone, who back 20 years ago actually built spool pieces at Argonne and shipped them out to Idaho to be put into the EBR-II circuit.

The spool piece consists of welding a 2-1/4 chrome schedule 30 pipe suitably tapered to match a 12-inch diameter schedule 10 stainless steel pipe, with the buttering technique using BP-85, which, vintage-wise, is the predecessor of Inconel 82, Inconel 132 and Inconel 182.

So the buttering technique was used on the 2-1/4 Cr-1Mo. They suitably stress relieved the joint, then machined the butter, and attached the stainless steel to the butter with the same material, Inconel.

They made three of these spools and shipped them out to the field. The spools were then field-welded in two of the nozzles in the EBR-II vessel, and everything has been fine for 19 years.

The superheater was taken out of service recently and is in pieces at Idaho. This joint is available and I suspect that someone will receive it and in the future, will look at it and see if there is anything interesting metallurgically; but I suspect that we'll find that it has served very well.

That's really all the one-to-one LMFBR reactor service that we can apply to Clinch River.

FOSSIL FUELED PLANT EXPERIENCE

So then what about the fossil business? The fossil plants operate at high temperatures and we have them all over the place. So let's look at the fossil experience. Transition joints in fossil plants have been around for more than 20 years. But recently the Steam Power Panel of the Metals Properties Council, Figure 12, convened a group to discuss the so-called problem, because there are failures in steam generators in fossil plants.

Representatives of the fabricators, the utilities, and the research laboratories met to address the cracking problem and to probably recommend some experiments and ultimately come up with some fixes which would prolong the longevity of fossil plant joints. The joints in these plants are about two inches in diameter with about a half-inch wall.

This type of joint is the reference joint with thousands of them in every boiler in contrast to what we're talking about at Clinch River or what I talked about previously. Keep in mind that the CRBRP spool pieces are 26 inches in diameter, about a half inch wall and about 18 inches in diameter and half inch wall.

To put the situation in perspective, once you get rid of the joints which are a consequence of sloppy workmanship or whatever, five years is the life exp tancy which most people plan on and do get; although the life expectancy runs to 150-, 200,000 hours for many joints.

But wouldn't it be nice if we could make them last the entire lifetime of the plant and with a high degree of reliability? So this group saw fit to create a panel to get a forum together where their problems could be discussed, conduct a survey amongst the utilities, and develop an initial experimental plan for, first, short-term thrust and subsequently to recommend a longer term plan for subsequent funding, which they have achieved. EPRI now has a long-term funded program which I will discuss later.

Figure 13 shows an old transition joint from a fossil plant. In 1977 we cut it up after it had been on line for 17 years. It is 2-1/4 Cr-1Mo to 132 (that's Inconel 82-A, 132, 182, same generic family) attached to 321 stainless steel, which is a favorite superheater material, too, for most boilers.

The joint lasted 17 years. It ran at a metal temperature of about 1125°F, although it made 1050 steam, for the most part. It had 146 up and down thermal cycles, at about the rate of about 200 degrees per hour. All this is reported in a paper by R. J. Gray, Oak Ridge, circa 1977.

Figure 14 shows another photomicrograph from Gray's paper. It defines the generic situation that we all expect. At the interface between the Inconel and the 2-1/4 Cr-1Mo, close to the fusion line, intergranular cracking and fissures are seen at the head of the crack.

You know, fissures move along. They open up and after an extended period of time (in this case 17 years, over 150,000 hours) there is a tendency for the formation of $M_{23}C_6$ (iron, chromium, nickel, molybdenum carbides) similar to sensitization precipitation at the fusion line.

Two observations can be made: carbides form and inter-granular cracking occurs with a fissure at the other end. That is what always happens at the interface between the weld metal and the 2-1/4 chrome 1 moly. So any tricks you can perform or any design that can enhance the situation at that interface is beneficial.

Now the first step that this panel did was to conduct a survey, Figure 15. They contacted 147 utilities and received 50 responses, which is a pretty good batting average.

Twenty utilities initially admitted to problems and, since then, more. The problem is not clearly defined. If it's an occasional failure, you fix it and quit worrying about it. But there have been instances of larger numbers of failures and there are a variety of reasons for failures, actually no single reason because welds are complex. Everybody has got his favorite joint and favorite joint design. The material selection is variable. The amount of strain associated with a particular boiler design or a particular plant also varies. The number of cycles varies. And the quality assurance varies. How seriously does the fabricator or the utility take it? Does he risk sloppy workmanship because it's an easy fix, or what?

In order to really have a long-term improvement to the situation, the panel decided what it needed is a consolidated research program, so they developed one.

The first thrust is shown in Figures 16, 17, and 18. TVA is one of the participants that volunteered to test the steam plant near my home in Tennessee to put a bunch of specimens in. They had a group of fabricators who built boilers make the specimens. Each fabricator made 45 specimens, 44 to go into the system, one to be saved as an archive. So every year or so during a scheduled down outage, they'll pull out a specimen and send it to metallography for examination.

The prediction is the story will go 44 years. Now, Combustion Engineering built joints of 2-1/4 chrome-1Mo to 316, with Inconel 82 (gas tungsten arc) used as a root bead. No backing rings were used. CE used Inco Rod A, which is an Inconel similar to the other ones you hear about. And that's at a 321 spool piece. And that becomes a sample that goes into TVA.

Foster Wheeler provided joints of 2-1/4 chrome-1Mo with a root bead of Inconel 82 with 182 stick electrodes (similar to A) as their candidate 45 joints.

The Boiler Tube of America Corporation, which probably has the nearest thing which resembles the Clinch River Joint design, which is described elsewhere, used 2-1/4 Cr-1Mo, a root bead of Inconel 82, and tungsten arc Inconel 82 as filler to a spool piece of Inconel 800 (and you'll see why that is appropriate a little later) then to 321 stainless steel with 16-8-2 tungsten arc.

B&W provided joints of 2-1/4 Cr-1Mo, Inconel 82 root, welded with 182 electrodes to 321 stainless steel. B&W and Foster Wheeler duplicated the exact joint.

An in-house fabricator, AEP (American Electric Power) Figure 18, built for their own plants samples which they are contributing to the program, approximately 40 joints. They elected to go from 2-1/4 Cr-1Mo to a spool piece of 800, with Inconel 82 at the root, and, in one case, (case 1), Inconel 82 all the way, then orbital welding. In case 2, AEP went from 2-1/4 Cr-1Mo to Inconel 82 root, Inco-rod A to alloy 800. In case 3, they went from 2-1/4 Cr-1Mo to 82, 182 stick electrode to 800. And then they take these pieces and shove them in the line with the same procedure they used in the shop to stick the pieces together.

So we do have four sets of joints (Figure 19). They are a triplex, if you want to call them that.

Now, you'll notice there was no 309, or stainless steel, which had been used for years as a joint material. It is commonly accepted that it is a bad choice because of the carbon migration into stainless steel versus nickel-base alloy, for one thing. The experience with 309 stainless steel is very much worse, as you will see a little later.

The point I want to make is that triplexes are used in many cases. The ones in the experimental program look just like the CRBRP joint, except for the welding processes, but the metallurgical structure is going to be the same. All the problems that have ever been seen occur right in the Inconel 2-1/4 Cr-1Mo weld, in the ferritic HAZ. So what happens elsewhere is important only in the sense that it affects what goes on in this weld. The gradual coefficient of expansion transition through this point surely, from a design standpoint, does help the stress situation at this point.

Therefore, one could maybe jump the gun and say a triplex ought to be better than a duplex, as I think is discussed elsewhere by our colleagues from General Electric on the project.

So those joints are on line. They've been on line about a year and a half at Kingston. Early life examinations will not show us much, so it may be five years before we see anything significant.

Figure 20 summarizes the statement made earlier that Inco 82 A and 182 and 132 are essentially high nickel alloys with chrome, manganese and columbium to make them into welding electrodes as a common vintage. So, when we talk about all these materials, we're essentially taking about Inconel-weld metal.

The committee then finally approached EPRI to fund the long-range program, Figure (21). As you can see, the program includes analysis of dissimilar weld failures, metallography, simulation tests, stress analysis, and development of guidelines for fabrication of improved welds.

In Figures 22 and 23 we see the key factors in failure.

Figure 24 points out the fact that the utilities are cooperating in the program. Many utilities have donated specimens for post-test examinations at General Atomics. The results of these examinations will be reported by the group when they become available.

Figure 25 details the CE Task 5 - Development of an Accelerated Test which is the one I like best, because it seems to be approaching the point where you can study a specimen and learn something about it. There are two main approaches. First, take the specimen and cycle it (a tubular specimen loaded with 9 ksi and cycle it up and down) at 1100 degrees Fahrenheit, and see what happens.

Well, the fact is that not much happens. Therefore, lets's do something worse. Put a bending load on another set of specimens. They put a little bending on it and increased the load another 12 ksi.

They will heat them for about 256 cycles which corresponds to twice a week for about 20,000 hours, and then take them out and do a detective job on them; metallography, mechanical testing, creep, notch toughness and so on. Some of the preliminary results are shown in Figure 26.

The GA rig is just essentially like the CE rig except much more sophisticated, much better instrumented, with thermocouples, with strain gauges, and with carefully recorded logs and thermal history; so, when they are in business, they'll have a lot more information for a stress analyst to analyze.

They have succeeded in their preliminary testing program to crack 309 stainless steel. That is as far as they have gone. I suspect that, in the future years, that they'll be doing the lion's share of the testing for the EPRI program, which is a three-year program.

This has been a review of the industrial experience with transition welds both within and outside of the nuclear field. I'm going to show you what is discussed in detail elsewhere, the Clinch River joint, Figure 27, which is 2-1/4 Cr-1Mo with the alloy 800 spool piece welded with Inconel 82 and with the 16-8-2 weld metal to the 316 stainless steel.

Remember that this triplex joint in CRBRP is preferable because of thermal expansion matching. Also, the difference between this joint and all the others I have discussed is the 950 degrees Fahrenheit maximum temperature service relative to the 1125. I talked about fossils, relative to the 820.

I considered the EBR, too, and the few cycles in the CRBRP, 30, 40, 50 versus the hundreds and two hundreds that one normally experiences in the fossil plant.

There's no question about the QA in CRBRP, namely, tender-loving care in manufacturing procedures is going to be used and has been used on the test articles for the Clinch River Plant versus the generally less stringent QA in the fossil plants.

There are only a few welds in the CRBRP, just a handful of welds, versus hundreds of thousands of welds in the fossil plants.

And the CRBRP temperature is 950. How long is it going to take for carbon diffusion? How long is it going to take for $M_{23}C_6$ to form? In a sense that joint takes advantage of everything we can think of and all the latest data available. In Figure 28, I just present the composition of all the materials we've been discussing for the CRBRP joints.

I could have stopped with my introduction. The bottom line is "We feel good about this CRBRP joint" and, as you read the other reports, I'm sure you'll come to the same conclusion I have, and I feel pretty good about it.

FIGURE 3.0-1

TRANSITION JOINT EXPERIENCE AND TECHNOLOGY

P. PATRIARCA AND G. M. GOODWIN

presented to

NUCLEAR REGULATORY COMMISSION
Bethesda, Maryland

April 6–7, 1982

OAK RIDGE NATIONAL LABORATORY
Oak Ridge, Tennessee 37830
operated by
UNION CARBIDE CORPORATION
for the
DEPARTMENT OF ENERGY

ORNL-DWG 82-4491 ETD

UNION CARBIDE ORNL

CRBRP UTILIZES THREE KEY MATERIALS IN PRESSURE BOUNDARY COMPONENTS

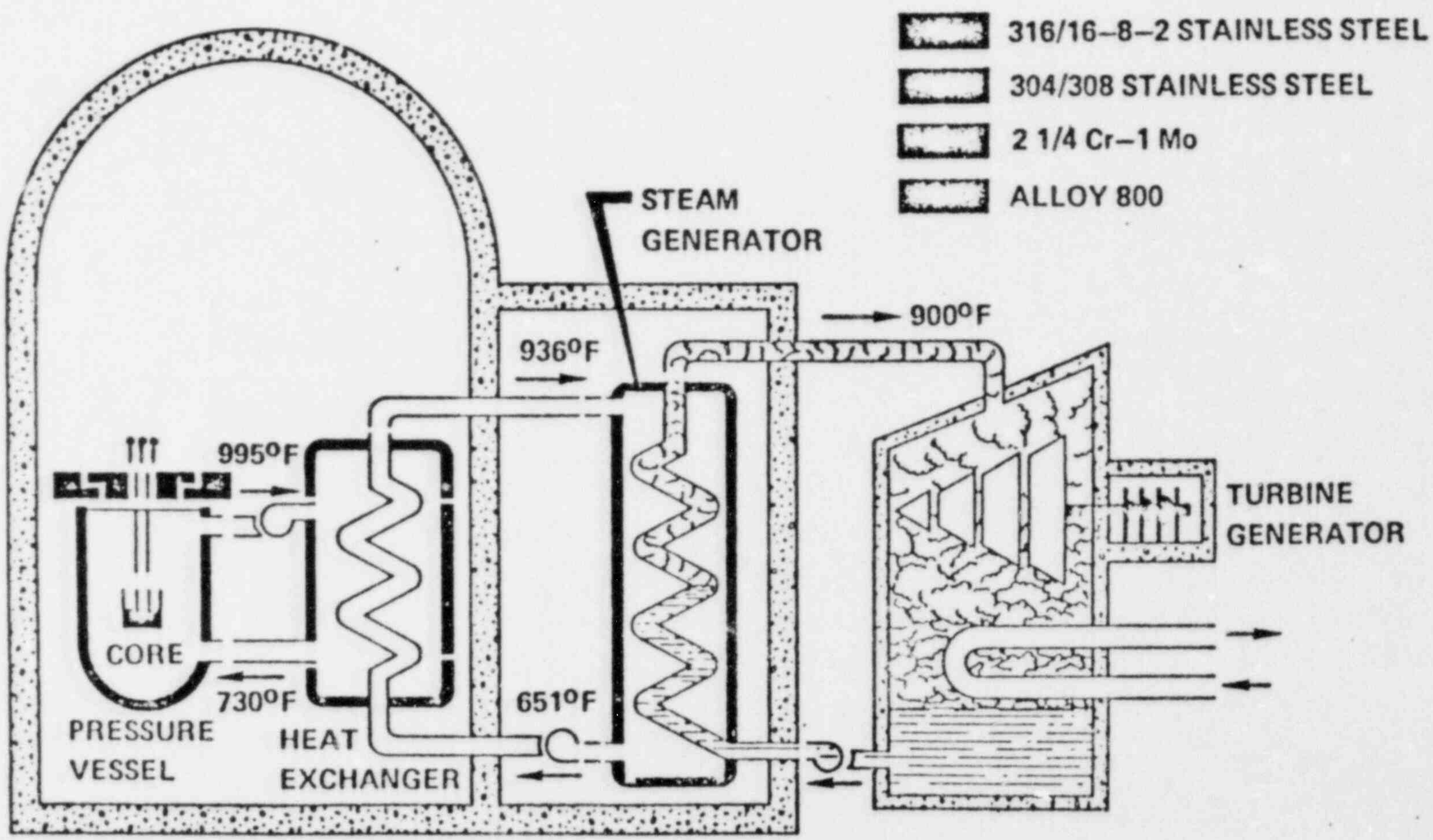


FIGURE 3.0-2

FIGURE 3.0-3

CRBRP HEAT TRANSPORT SYSTEM MATERIALS
AND STRUCTURES

TRANSITION JOINT EXPERIENCE

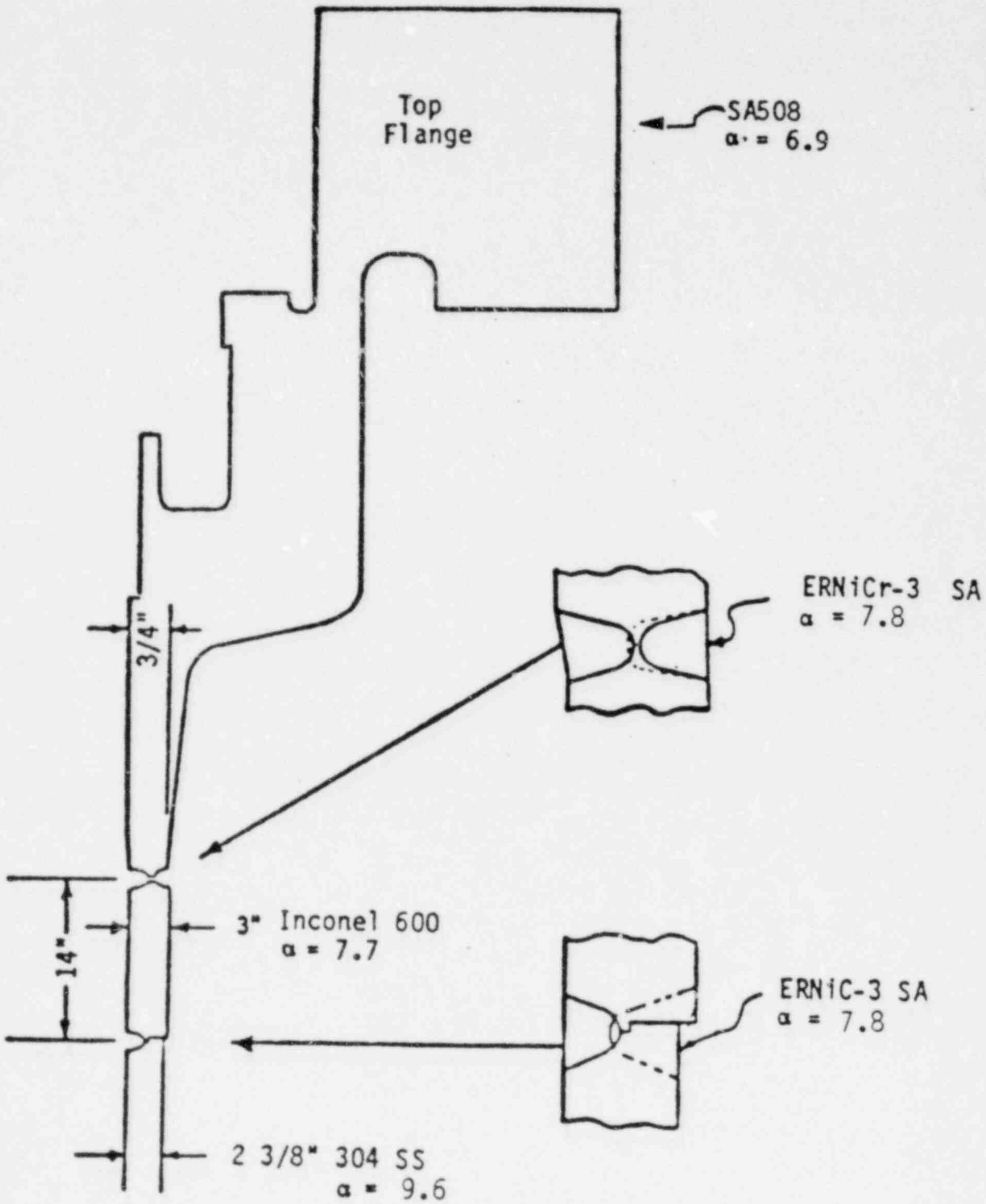
- PURPOSE
  - TO ESTABLISH AWARENESS OF SERVICE EXPERIENCE WITH TRANSITION JOINTS

- SCOPE: FOSSIL, LMFBR, PWR
  - JOINT DESIGNS
  - SERVICE CONDITIONS
  - PERFORMANCE

- CONCLUSION
  - PRIOR EXPERIENCE PROVIDES SUFFICIENT UNDERSTANDING TO ENABLE CREATION/EVALUATION OF THE CRBRP TRANSITION JOINTS

FIGURE 3.0-4

REACTOR VESSEL TRANSITION JOINT



α = CTE (70–450°F)(10^{-6} in./in./°F)

FIGURE 3.0-5

TYPICAL PWR, BWR PRIMARY VESSEL "SAFE END" WELD

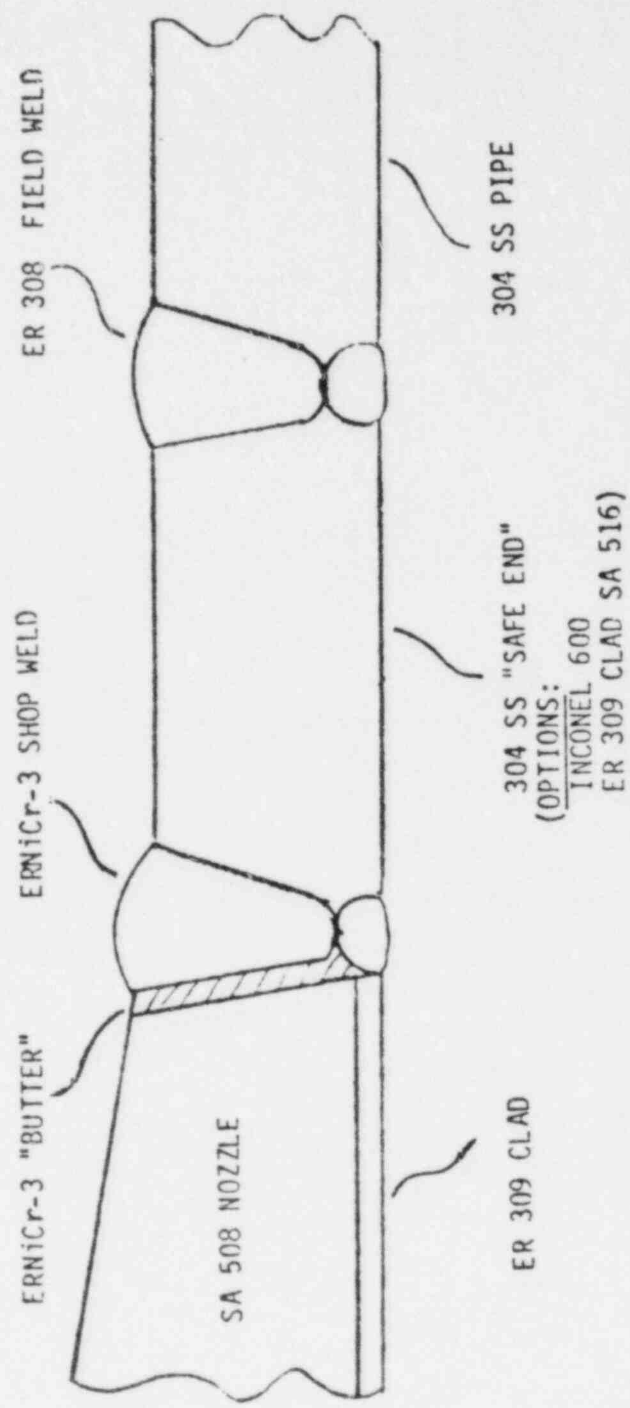


FIGURE 3.0-6

TRANSITION JOINTS IN PWR PLANTS IN OPERATION
IN EXCESS OF TEN YEARS

Unit	MW(e)	Nuclear Supplier	Commercial Operation (Month/Year)
Shippingport	150	Westinghouse	12/1957
Yankee-Rowe	175	Westinghouse	7/1961
San Onofre-1	430	Westinghouse	1/1968
Connecticut Yankee	582	Westinghouse	1/1968
Ginna	490	Westinghouse	7/1970
Point Beach-1	497	Westinghouse	12/1970
Robinson-2	665	Westinghouse	3/1971
Point Beach-2	497	Westinghouse	10/1972
Surry-1	788	Westinghouse	12/1972
Maine Yankee	825	Combustion Engineering	12/1972
Turkey Point-3	728	Westinghouse	12/1972

FIGURE 3.0-7

TRANSITION JOINTS IN BWR PLANTS IN OPERATION
IN EXCESS OF TEN YEARS

Unit	MW(e)	Nuclear Supplier	Commercial Operation (Month/Year)
Dresden-1	207	General Electric	8/1960
Big Rock Point	71	General Electric	12/1965
Genoa-2	48	Allis-Chalmers	11/1969
Oyster Creek	620	General Electric	12/1969
Nine Mile Point-1	610	General Electric	12/1969
Dresden-2	794	General Electric	7/1970
Millstone-1	652	General Electric	12/1970
Monticello	536	General Electric	6/1971
Dresden-3	794	General Electric	10/1971
Pilgrim-1	655	General Electric	7/1972
Quad Cities-1	789	General Electric	8/1972
Vermont Yankee	514	General Electric	12/1972

FIGURE 3.0-8

NOMINAL CHEMISTRY OF MATERIALS IN CRBRP PRIMARY VESSEL TRANSITION JOINTS

Material	Composition, %							α^*
	Ni	Cr	Fe	C	Mo	Mn	P	
Type 304 SS	9.5	18.5	Bal	0.05	—	2.0	0.030	9.6
ERNiCr-3	72.5	20.0	1.05	0.012	0.03	3.11	0.003	7.8
Inconel 600	Bal	15.8	7.2	0.04	—	0.2	—	7.7
ERNiCr-3	72.5	20.0	1.05	0.012	0.03	3.11	0.003	7.8
A508Cl 2	0.75	0.35	Bal	0.27	0.6	0.75	0.025	6.9

* α = CTE (70–450°F)(10^{-6} in./in./°F).

IHTS TRANSITION JOINTS

LOCATIONS

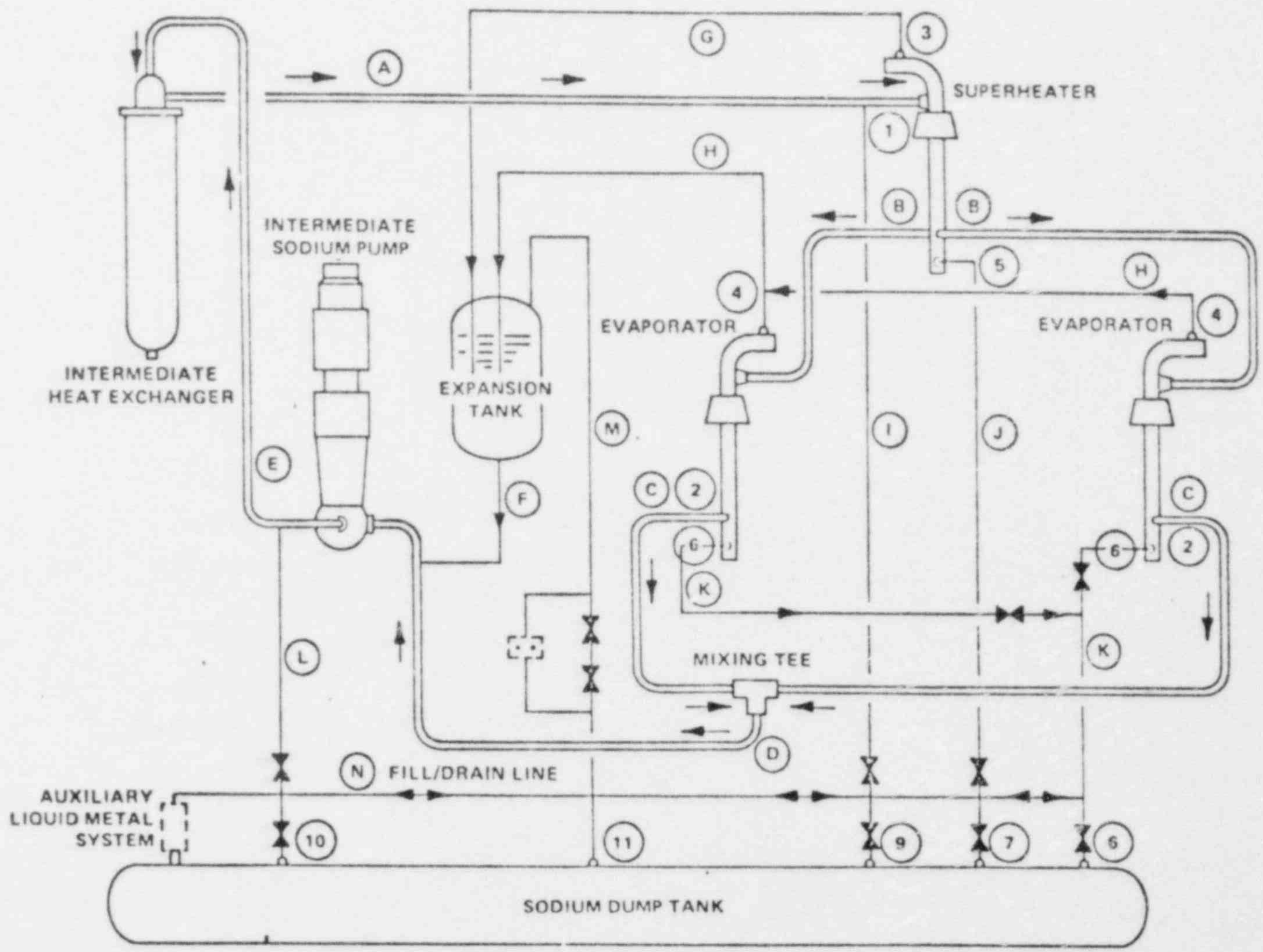


FIGURE 3.0-9

TRANSITION JOINTS IN LMFBR DEMONSTRATION PLANT SUPERHEATERS
OPERATING IN EXCESS OF TEN YEARS

Parameters	EFAPP (US)	EBR-II (US)	BN-350 (USSR)	SNR-300 (FRG)
Years Operated	1963–1973	1963–Present	1973–Present	1983 Criticality
MW(t)	200	62	1000	762
MW(e)	60	19	150*	312
Type of Steam Generator Unit	Once-through single wall involute	Recirculating duplex tube	Shell & Bayonet Single Wall	Once-through 2 loops straight tube single wall 1 loop helical
Number of Units/Plant	3	1	12	9
Superheat Steam Temperature, °C(°F)	416(780)	438(820)	435(815)	495(920)
Steam Pressure, MPa (psig)	6.21(900)	8.62(1250)	5.67(735)	1.59(2300)
Sodium Inlet Temperature, °C(°F)	438(820)	465(870)	450(842)	526(980)
Tube Material	2 1/4 Cr-1 Mo	2 1/4 Cr-1 Mo	2 1/4 Cr-1 Mo	2 1/4 Cr-1 Mo-Nb

*BN-350 – Balance for Desalting.

FIGURE 3.0-10

EBR-II TRANSITION JOINT

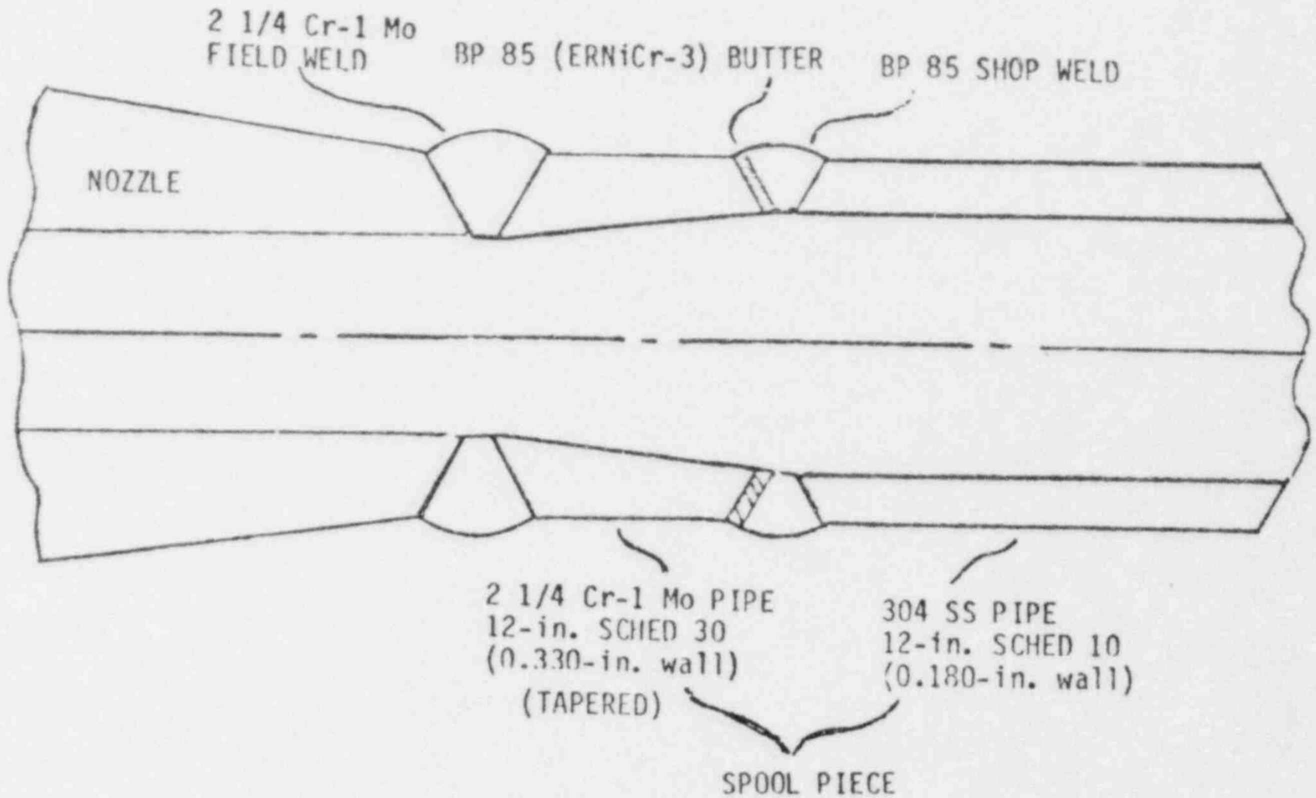


FIGURE 3.0-11

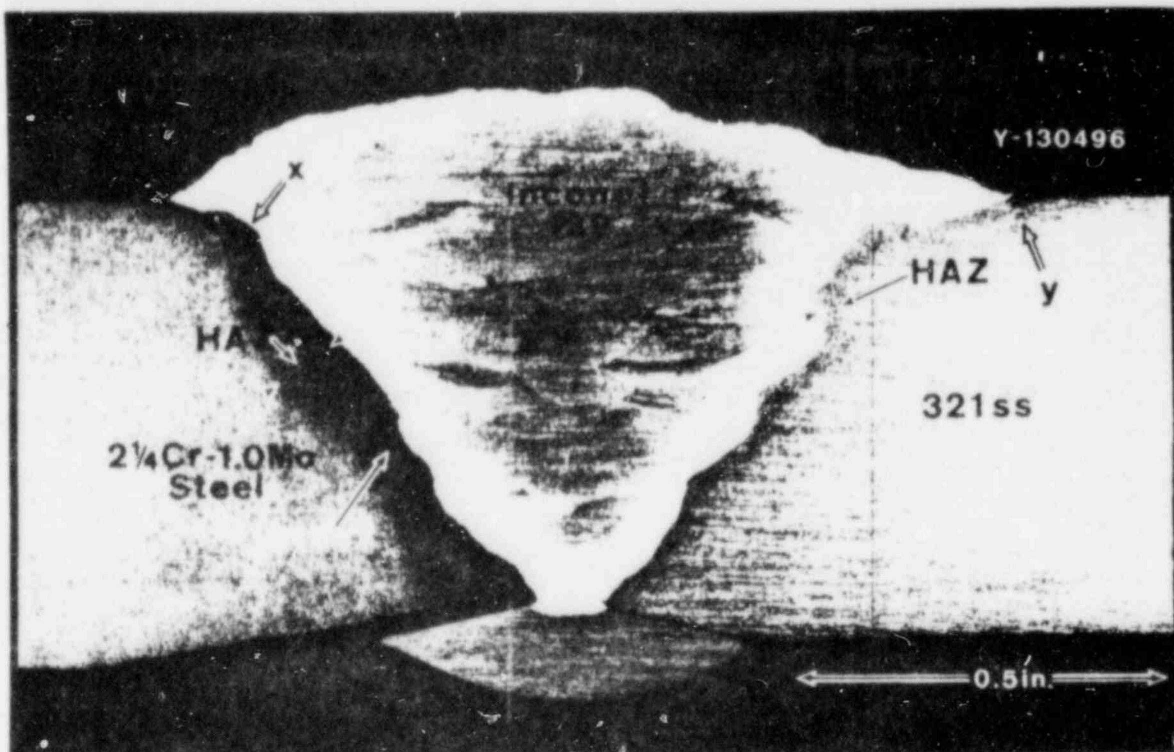
FIGURE 3.0-12

ORNL WS-15376

DISSIMILAR METAL WELD TASK GROUP

IN 1977, THE STEAM POWER PANEL FORMED A SPECIAL TASK GROUP. ITS CHARTER WAS TO "ASSESS THE SIGNIFICANCE OF THE DISSIMILAR WELD INTERFACE CRACKING PROBLEM AND RECOMMEND EXPERIMENTAL PROGRAMS TO UNDERSTAND THE PHENOMENON AND PRODUCE AN ULTIMATE LONG-TERM FIX." THE ACTIVITIES OF THE TASK GROUP HAVE CENTERED ABOUT THE FOLLOWING AREAS:

- FORUM FOR DISCUSSIONS AMONG UTILITIES, BOILER MANUFACTURERS, SUPPLIERS, RESEARCH LABORATORIES, UNIVERSITIES, AND OTHER INTERESTED ORGANIZATIONS
- SURVEY TO DETERMINE SEVERITY OF PROBLEM
- EXPERIMENTAL JOINT EVALUATIONS (SHORT-TERM)
- LONG-RANGE EXPERIMENTAL PROGRAM

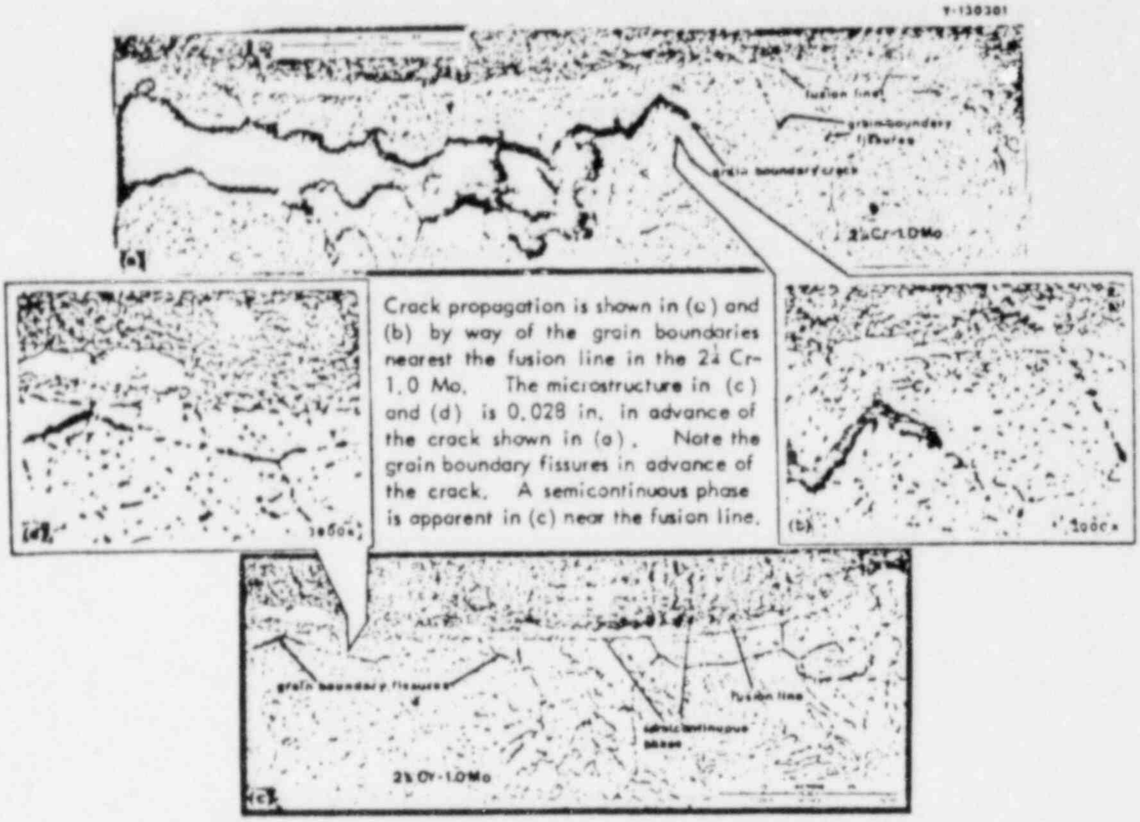


LONGITUDINAL CROSS SECTION OF TRANSITION JOINT
Areas designated "x" and "y" receive specific attention in this examination. Heat Affected Zones (HAZ) in the 2 1/4 Cr-1.0 Mo steel and type 321 stainless steel are indicated.

BABCOCK & WILCOX — FOSSIL PLANT — 17 YEARS SERVICE

Mean Metal Temperature, 1125°F
146 Cycles (200°F/h Cooling Rate)

FIGURE 3.0-13



Crack propagation is shown in (a) and (b) by way of the grain boundaries nearest the fusion line in the 2¼ Cr-1.0 Mo. The microstructure in (c) and (d) is 0.028 in. in advance of the crack shown in (a). Note the grain boundary fissures in advance of the crack. A semicontinuous phase is apparent in (c) near the fusion line.

FAILED TRANSITION JOINT : CRACK PROPAGATION

FIGURE 3.0-14

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SURVEY

UTILITIES ASKED	147
UTILITIES RESPONDING	54
UTILITIES EXPERIENCING DISSIMILAR METAL WELD FAILURES	20

CONCLUSIONS:

- PERFORMANCE IS INCONSISTENT
- REASONS FOR MARKED DIFFERENCES ARE NOT KNOWN BECAUSE THE WELDS ARE A VERY COMPLEX STRUCTURE WITH SIGNIFICANT VARIATIONS IN STRAIN, MATERIAL PROPERTIES, AND OPERATING CONDITIONS
- A RESEARCH PROGRAM SHOULD BE INITIATED TO UNDERSTAND AND SOLVE THE PROBLEM

FIGURE 3.0-15

ORNL WS-15379R

EXPERIMENTAL JOINT EVALUATION PROGRAM (TVA PORTION)

45 DUTCHMEN WERE FABRICATED BY EAC CONTRIBUTING COMPANY:

- 44 INSTALLED IN TVA'S KINGSTON STEAM PLANT (~ 1/yr OF SERVICE)
- 1 RETAINED BY TVA (WITH RECORDS) AS A BASELINE STANDARD)

COMBUSTION ENGINEERING

- 2¼ Cr–1 Mo STEEL TO 321 H STAINLESS STEEL WITH INCONEL [GTA (INCONEL 82) FOR ROOT PASS, SMA INCO WELD-A]

FOSTER WHEELER

- 2¼ Cr–1 Mo STEEL FOR 321 H STAINLESS STEEL WITH INCONEL [GTA (INCONEL 82) FOR ROOT; SMA (INCONEL 182) FOR FILL PASSES]

FIGURE 3.0-16

ORNL WS-15380R

EXPERIMENTAL JOINT EVALUATION PROGRAM
(TVA PORTION)
(CONTINUED)

- BOILER TUBE COMPANY OF AMERICA
  - 2¼ Cr–1 Mo STEEL TO INCOLOY 800 H WITH INCONEL 82 (GTA FOR ALL PASSES)
  - INCOLOY 800 H TO 321 H STAINLESS STEEL WITH 16-8-2 STAINLESS STEEL (GTA FOR ALL PASSES)
- BABCOCK AND WILCOX
  - 2¼ Cr–1 Mo STEEL TO 321 H STAINLESS STEEL WITH INCONEL [GTA (INCONEL 82) ROOT PASS; SMA (INCONEL 182) FILL PASSES]

FIGURE 3.0-17

ORNL WS-15381

EXPERIMENTAL JOINT EVALUATION PROGRAM (AEP PORTION)

AEP FABRICATED THE FOLLOWING WELDS (8 WELDS PER PLATEN) FOR INSTALLATION JUST ABOVE THE ROOF ON THE REHEATER OUTLET BANK

ALL WELDS JOIN 2¼ Cr–1 Mo STEEL SAFE ENDS TO INCOLOY 800 H TUBES

- 5 "A" PLATENS WITH ORBITAL GTA WITH INCONEL 82
- 5 "B" PLATENS WITH GTA ROOT (INCONEL 82) AND SMA FILL PASSES (INCONEL A)
- 5 "C" PLATENS WITH GTA ROOT (INCONEL 82) AND SMA FILL PASSES (INCONEL 182)

FIGURE 3.0-18

EXPERIMENTAL PROGRAM — DUTCHMEN TUBE AND WELD MATERIALS

Supplier/Number of Dutchmen	"Lo Temp" Superheater	Root Pass	Filler Pass	Spool-Piece	Root Pass	Filler Pass	"High Temp" Superheater
			TVA STEAM PLANT DUTCHMEN				
CE/45	2 1/4 Cr-1 Mo	82 GTA	A SMA	—	—	—	321 SS
FW/45	2 1/4 Cr-1 Mo	82 GTA	182 SMA	—	—	—	321 SS
BTA/45	2 1/4 Cr-1 Mo	82 GTA	82 GTA	800	16-8-2 GTA	16-8-2 GTA	321 SS
B&W/45	2 1/4 Cr-1 Mo	82 GTA	182 SMA	—	—	—	321 SS
			AEP STEAM PLANT DUTCHMEN				
AEP/40	2 1/4 Cr-1 Mo	82 GTA	82 GTA	800			321 SS
AEP/40	2 1/4 Cr-1 Mo	82 GTA	A SMA	800	Field Welds[a]		321 SS
AEP/40	2 1/4 Cr-1 Mo	82 GTA	182 SMA	800			321 SS

[a]Inconel weld metals as in shop welds.

FIGURE 3.0-19

NOMINAL CHEMISTRY OF MATERIALS IN TVA AND AEP DUTCHMEN

	Composition, %								
Material	Ni	Cr	Fe	C	Mo	Mn	P	S	Nb
2 1/4 Cr-1 Mo	—	2.20	Bal	0.11	0.95	0.49	0.011	0.030	—
Inconel 82	72.5	20.0	1.05	0.012	0.03	3.11	0.004	0.003	2.25
Inconel 182	67.0	14.0	7.5	0.05	—	7.75	0.004	0.008	1.75
Inconel A	70.0	15.0	9.0	0.03	1.5	2.00	0.004	0.008	2.00
Incoloy 800	33.0	21.5	Bal	0.08	—	1.22	0.008	0.001	—
16-8-2 SS	9.0	16.2	Bal	0.06	—	1.5	0.015	0.02	—
321 SS	9.5	18.5	Bal	0.08	—	2.0	0.030	0.010	5XC(Ti)

FIGURE 3.0-20

ORNL WS-15383

LONG-RANGE PROGRAM

TASK 1 ANALYSIS OF CAUSES OF DISSIMILAR WELD FAILURE
- SUBTASK 1 DATA/EXPERIENCE SURVEY
- SUBTASK 2 METALLURGICAL AND MECHANICAL TESTING OF SERVICE WELDS

TASK 2 ANALYSIS OF DISSIMILAR WELDS WITH FOSSIL FIRED BOILER SERVICE

TASK 3 SIMULATION OF METALLURGICAL CHANGES THAT OCCUR IN DISSIMILAR WELDS DURING SERVICE

TASK 4 STRESS ANALYSIS OF DISSIMILAR WELDS
- SUBTASK 1 ASSESSMENT OF CURRENT TECHNIQUES
- SUBTASK 2 MATERIAL PROPERTIES

TASK 5 DEVELOPMENT OF AN ACCELERATED LABORATORY TEST

TASK 6 DEVELOPMENT OF GUIDELINES FOR FABRICATION OF IMPROVED DISSIMILAR WELDS

FIGURE 3.0-21

ORNL WS-15384

TASK 1/2 PROGRESS

- UNPUBLISHED AND PUBLISHED DOCUMENT/EXPERIENCE REVIEW 50% COMPLETE. APPRECIABLE AMOUNTS OF UNPUBLISHED DATA OBTAINED FROM UTILITIES AND FABRICATORS
- DOCUMENTS CONFIRM THAT KEY FACTORS IN FAILURE ARE:
  - FILLER METAL COMPOSITION (NICKEL BASE CONSIDERED SUPERIOR, BUT NOT IMMUNE)
  - EXPANSION/CONDUCTIVITY MISMATCH
  - OXIDATION 'NOTCHING' (SOME BELIEVE OXYGEN ESSENTIAL FOR FAILURE)
  - CARBON MIGRATION (AGGRAVATES STRENGTH MISMATCH)
  - TYPE OF LOADING INCLUDING CYCLIC AND BENDING LOADS
  - POST WELD HEAT TREATMENT (EFFECTS CONTROVERSIAL)

FIGURE 3.0-22

ORNL WS-15385

TASK 1/2 PROGRESS (CONTINUED)

- SERVICE DATA INDICATES THAT FAILURES ARE MOST COMMON:
  - IN HIGHER TEMPERATURE REGIONS
  - IN TUBES GREATER THAN 2 in. O.D. IN SUBCRITICAL BOILERS (IN FACT NO FAILURES SO FAR FOUND IN TUBES LESS THAN 2 in. O.D. IN SUBCRITICAL BOILERS)
  - IN SUPERCRITICAL PLANTS (ALSO SHORT FAILURE TIMES)
  - IN PLANTS WITH HIGHER NUMBER OF CYCLES ACCUMULATED
  - IN DESIGNS WHERE T-22 THICKNESS CHANGE MADE AT JOINT
- SERVICE WELDMENTS FOR METALLURGICAL AND MECHANICAL TESTING:
  - DOCUMENTATION OF SERVICE HISTORY DISCUSSED WITH UTILITIES
  - TENTATIVE COMMITMENTS OBTAINED FOR MOST WELDMENT TYPES
  - ADDITIONAL, COMPARATIVE METALLURGICAL SAMPLES LOCATED

FIGURE 3.0-23

ORNL WS-15389

SERVICE WELDMENT SAMPLES FOR DISSIMILAR WELD FAILURE ANALYSIS AND DEVELOPMENT
TASK 1 AND TASK 2

COMBINATION	TYPE OF SAMPLE	SERVICE	LOCATION	CONDITION	POTENTIAL SOURCES
1	T-22/309/300H SERIES S/S	100,000 h	HOR. S/H	SOUND	DETROIT EDISON
	"	"	"	CRACKED	TENNESSEE VALLEY AUTHORITY
2	"	"	PEND. S/H	SOUND	UNITED ILLUMINATING CO.
					DETROIT EDISON
	"	"	"	CRACKED	PENNSYLVANIA POWER AND LIGHT CO.
					PENNSYLVANIA ELECTRIC CO.
3	T-22/NICKEL/300H SERIES S/S BASE	50,000	HOR. S/H OR PEND. S/H	SOUND	AMERICAN ELECTRIC POWER SERVICE CORP.
					TENNESSEE VALLEY AUTHORITY
4	"	100,000	HOR. S/H	SOUND	AMERICAN ELECTRIC POWER SERVICE CORP.
	"	"	"	CRACKED	
5	"	"	PEND. S/H	SOUND	TENNESSEE VALLEY AUTHORITY
	"	"	"	CRACKED	
6	"	150,000	HOR. S/H OR PEND. S/H	SOUND	DETROIT EDISON
					TENNESSEE VALLEY AUTHORITY
					AMERICAN ELECTRIC POWER SERVICE CORP.
7	T-22 BASE-METAL	100,000 h	HOR. S/H	SOUND	DETROIT EDISON
					TENNESSEE VALLEY AUTHORITY
					AMERICAN ELECTRIC POWER SERVICE CORP.

FIGURE 3.0-24

ORNL WS-15387

COMBUSTION ENGINEERING
TASK 5 – DEVELOPMENT OF AN ACCELERATED TEST

- CYCLIC TUBULAR RUPTURE
  - CYCLE TO 1100°F AND 9 ksi
- EXTERNAL BENDING MOMENT
  - CYCLE TO 1100°F AND 12 ksi TOTAL AXIAL STRESS PRODUCED BY BENDING AND INTERNAL PRESSURE
- LONGITUDINAL WELDS
- EVALUATION OF SERVICE SIMULATION SPECIMENS (SPECIMENS REMOVED FROM CYCLIC TUBULAR RUPTURE FACILITY AFTER 20,000 h AND 256 cycles)
  - OPTICAL AND SCANNING MICROSCOPY
  - UNIAXIAL CREEP RUPTURE
  - SHEAR CREEP RUPTURE
  - LOW TEMPERATURE FRACTURE

FIGURE 3.0-25

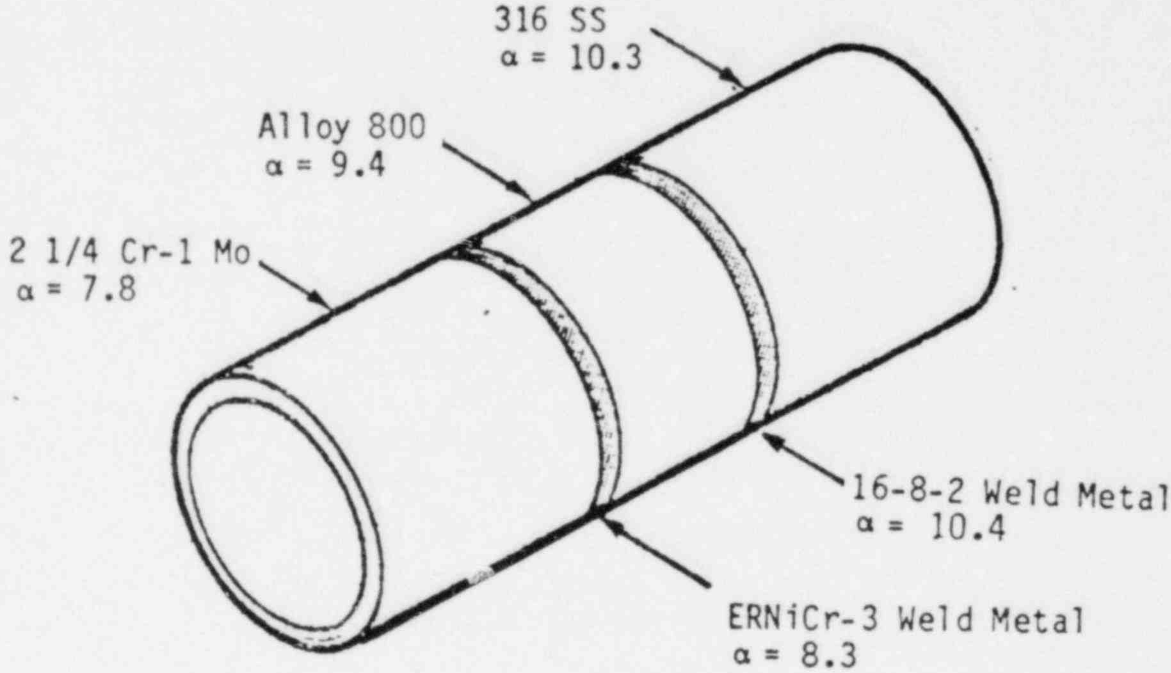
ORNL WS-15388

COMBUSTION ENGINEERING
TASK 5 – DEVELOPMENT OF AN ACCELERATED TEST
SOME PRELIMINARY RESULTS

- CYCLIC TUBULAR RUPTURE
  - PRODUCED FAILURES TYPICAL OF SERVICE FOR 309 AND INDUCTION PRESSURE WELDS BUT NO FAILURES AFTER 360 CYCLES AND 23,000 h FOR NICKEL-BASE OR SPOOLPIECE JOINTS (ALLOY 800, IN 102, AND INCONEL 625)
- EXTERNAL BENDING MOMENT – NO FAILURES AFTER 3000 h AND 20 BENDING CYCLES FOR 309, INCONEL 132 AND INCONEL 82 JOINTS
- LONGITUDINAL WELDS
  - FAILURE TIMES FROM 3000 TO 8000 h FOR 309, INCONEL 182 AND INCO A-WELDS. CRACKING FOR NICKEL WELDS WAS ALONG $M_{23}C_6$ PARTICLES AND INTERGRANULAR IN FERRITIC HAZ

FIGURE 3.0-26

TRI-METALLIC TRANSITION JOINT DESIGN



α = CTE (70–1000°F)(10^{-6} in./in./°F)

FIGURE 3.0-27

NOMINAL CHEMISTRY OF MATERIALS IN CRBRP STEAM GENERATOR TRANSITION JOINTS

Material	Composition, %								α*
	Ni	Cr	Fe	C	Mo	Mn	P	S	
2 1/4 Cr-1 Mo	—	2.20	Bal	0.11	0.95	0.49	0.011	0.030	7.8
ERNiCr-3	72.5	20.0	1.05	0.012	0.03	3.11	0.004	0.003	8.3
Incoloy 800	33.0	21.5	Bal	0.08	—	1.22	0.008	0.001	9.4
ER16-8-2	9.0	16.2	Bal	0.064	—	1.45	0.015	0.019	10.4
316H SS	12.5	17.5	Bal	0.05	2.30	1.9	0.030	0.010	10.3
304 SS	9.5	18.5	Bal	0.05	—	2.0	0.030	0.010	10.3

* α = CTE (70–1000°F)(10^{-6} in./in./°F)

FIGURE 3.0-20

REACTOR VESSEL TRANSITION JOINT

By

L. France (W-AESD) - Part 1

G. Nickodemus (W-ARD) - Part 2

REACTOR VESSEL JOINT - PART 1

The reactor vessel transition joint is in a low temperature region of the plant. We will discuss the geometry, material fabrications, the conditions under which it serves, and the evaluation that has been made of that particular joint. The conclusion that is reached in the presentation is that the design of the joint, given the service conditions, is such that we can be assured that the reactor vessel transition joint will maintain its integrity through the life of the plant.

There are two NRC questions on the record in this area, question 210.3 and question 250.5. We will be addressing those questions from the standpoint of the reactor vessel transition joint in this presentation, which discusses the geometry, location, material selection, fabrication, and operating environmental conditions, Figure 1. The following paper by Glen Nickodemus will cover the structural evaluation and analysis.

The location of this joint is in the upper part of the reactor vessel, approximately 49 inches down from this flange seat which supports the vessel (Figure 2). It is located in the cover gas area, above the sodium level.

The cross-section of the joint, Figure 3, illustrates the SA508 forging and the joint which is approximately 49 inches below the flange. It is in reality a tri-metallic joint, in that it includes an Inconel 600 (SB168) transition spoolpiece. And the bottom remainder of the vessel is SA-204 type 304 stainless steel. Both welds shown use Inconel 82 (ERNiCr-3) filler metal.

The design, materials and fabrication were all in accordance with the 1974 ASME Code, including the '74 winter addendum, and specified code cases.

Most of the code cases that were specified were basically for high temperature namely 1492, 3, 4, and really aren't applicable to this region of the vessel since it operates at low temperatures.

In all cases the code was supplemented by mandatory RDT standards, the main standard being E15-2NB, Class 1 nuclear components. This code invokes all the

Materials, M, standards; SA-508 Class 2 (M2-7), SB-162 (M5-4) for the Inconel 600, SA-240 Type 304 (M5-1), and SFA-514 (M1-11).

And it also invokes the non-destructive (F3-6) and welding and brazing qualifications (F6-5), supplements to the code. In all cases, any conflict between the RDT standards and the code, the code takes priority. We don't compromise the code as a result of using RDT Standards.

The fabrication sequence used to form the first (upper) joint which is 508 to the Inconel 600, is shown in Figure 5.

The first step was the fit-up to the machined SA508 forging. The Inconel 600 shell was then fitted to it and the ID was submerged-arc welded using Inconel 82 fillers.

It was then backgrooved and ground to remove at least an eighth of an inch minimum material from the zone affected by the backgrooving. The backgroove was PT'd and then the O.D. welded by a similar process, using the 82 filler and Incoflux 4. That weld materials combination was used for all welds.

Preliminary stress relief was given at that point in fabrication, 15 minutes at 1125°F. This was mainly to avoid any possibility of any delayed cracking as a result of possible hydrogen in the material. This was a waiver of the RDT standard requirement that preheat be maintained until the final heat treatment. It was impractical for this situation to maintain preheat for the time required to finish the weld.

At that point after the preliminary heat treatment, the weld surfaces were ground. The dye-penetrant test of the I.D. and the O.D. was performed and preliminary X-ray was performed on this joint, one normal and one angle shot using wide-spectrum X-ray to give better resolution of potential defects.

Then it was given final stress relief for ten hours at 1124°F, which answers an earlier question as to what was done for post-weld heat treatment. I think that code-wise that's an acceptable temperature. It wasn't intentionally pushed to a higher temperature. From a code standpoint that is acceptable.

To complete the upper joint it was given a final PT of both sides of the upper weld and a final RT of this weld, since RDT standard requires that the final inspection, the volumetric and surface dye penetrant be performed in the final heat-treated and final machined condition.

To complete the total transition joint, the bottom weld joint was fabricated by the sequence given in Figure 6. The 304 shell was fitted to the top by a ship-lap joint. The I.D. weld was again made by sub-arc process using Inconel 82 and Incoflux 4.

The backgroove was machined and welded, and then the I.D. and O.D. were ground.

The PT of the I.D. and O.D. were performed, and the final RT was performed using Iridium 192 as the isotopic source. This is a ground, smooth taper in the finished geometry of the vessel.

There was no other examination of the joint. Let me regress a little. In selecting weld materials and the materials used in the transition joints, we went through a process that is very similar to what has been described by Pete Patriarca in the previous paper. And that was a rather thorough review of the transition weld joint technology and performance.

We based the location, design and material selection for the ferritic to austenitic transition joint such that we mitigated the failure mechanisms and environmental conditions that basically were associated with the fossil fuel performance, in the 1100 to 1200 degrees Fahrenheit region, see Figure 7.

We considered the thermal environment and cyclic loading and problems that were related to failures of a higher temperature fossil fuel, the oxidation resistance problem, the carbon migration from the ferritic material, and the metallurgical stability and possible deterioration caused by the high temperature exposure.

Figure 8 provides a list of the potential problems and now I will describe why they are not problems or how we handled them.

The thermal environment, for instance, is a very low temperature one, 450 degrees Fahrenheit, for the ferritic joint. The lower one is around 600°F. It was intentionally located here to minimize, number one, the temperature; and number two, to minimize the magnitude of any thermal expansion or temperature response during transients in the reactor.

The environment that this joint sees is very benign. On the I.D. it sees argon and either sodium vapor or a sodium film, and at very low temperatures in comparison to what the remainder of the vessel is exposed to.

We used the nickel-base filler, Inconel 82, to minimize carbon migration both during welding and during the post weld heat treatment cycle. And again in the case of the carbon migration question, what comes into play here are the simple laws of diffusion. At the operating temperature for the classical fossil fuel plants, the temperatures which are encountered are around 1100°F, where the diffusion coefficient is about 10^{-9} centimeters square per second, and at 450°F approximately 10^{-16} centimeters square per second. To get the 10^{-16}, one is faced with extrapolations of available data by approximately three or four orders of magnitude. The process which I'm discussing here is the particle growth rate which is proportional to the square root of DT.

People just don't measure coefficients at the temperatures that we're operating this joint at. If you go through this exercise, you come up with a number that compares the two. The diameter growth rate is different, lower, by some 3,000 times or something in that order of magnitude for the low temperature of the reactor vessel joint.

As far as metallurgical stability is concerned, it's very difficult to find people who look for metallurgical instability at 450 degrees Fahrenheit. Basically, the phases that are present after the post-weld heat treatment will not change, i.e., in the lifetime of this plant one would not predict that $M_{23}C_6$ would be expected to form.

It has been stated that some experience has shown that the residual stresses are the prime deteriorating process in the bimetallic or trimetallic joints. I have made no estimate of the residual stresses, but I would assume they are approximately equal to the yield strength of the two materials at 1125°F. That's a good rule of thumb. It's hard to get them below that unless substantially higher post weld heat treatment temperatures are used.

The first four I have discussed are basically conditions that have always been considered in the way one approaches bimetallic joints. I'd like to look at the other conditions that this joint sees, Figure 9. It should be noted that I'm addressing only the ferritic joint. We really don't consider the austenitic joint to be a problem. This has never been shown to be a problem in fossile fuel applications and we don't consider it a problem here.

With regard to the radiation effects, the total fluence is less than 10^{17} neutrons per centimeter square (at E greater than 1 Mev) at end of life. In fact, that is probably two orders of magnitude high for the total fluence for the area. It is well below the threshold for any mechanical degradation as a result of the radiation in either of the materials. Surveillance is not required per 10CFR50, Appendix H.

Decarburization or corrosion due to the sodium film where the temperature is low is discussed by Bill Ray elsewhere. And at the temperature of 450°F there is no interstitial transfer or corrosion detectable in these materials due to the sodium.

As to nitriding, the external environment is essentially nitrogen with a controlled oxygen addition. And at temperatures below 600°F with any ferritic material it's impossible to get gas-phase nitriding.

Another corrosion mechanism that one might be concerned with which Patriarca mentions in his discussion is the problem of early safe end work where this piece was type 304SS and it was sensitized during furnace stress relief. We have precluded this by using the Inconel 600. So there is no intergranular attack possibility during storage or erection of the plant and other kinds of corrosion that you can come up with, galvanic or whatever are not present during operation because of the benign nature of that environment.

Basically the operating environment really precludes those. The I.D. and O.D. are both very benign environments for corrosion mechanisms that one normally considers.

Figure 10 summarizes the low temperature information given by Patriarca. He gave all the numbers and lifetimes and what have you regarding PWR experience. The material combination in the PWR safe ends, mainly the SA508 nozzle, Inconel 82 as a filler, to a type 304 wrought or cast type structure, have shown no service-related failures.

Their experience is quite good. My basic conclusions are, Figure 11, number one, that our analysis of this joint indicates that the reactor vessel materials will provide safe, reliable performance. Number two, material degredation is not anticipated due to the service temperature coupled with these benign environments which were discussed. And number three, nuclear experience confirms the adequacy of this particular material combination for the intended service temperature.

Thermal gradients are considered by Glen Nickodemus in a little more detail in the next discussion. Let me just go back to Figure 2 and point something out, that the joint which I'm discussing here is in the cover gas region. The sodium level is so far below the joint that, in reality, in the life of this joint it will never see direct contact with the coolant. You would expect very slow transient response in relation to what you would normally see elsewhere in the plant in relation to an upset or something like this. What you're really looking at here is cover gas environment with at most a condensed sodium film on it.

In answer to the question of whether there is any accident that could put hot sodium in contact with the weld, I know of none that would bring sodium up that high. We have looked at the potential for the sodium level to change there. And we have provided specifically for appropriate cover gas control to maintain the pressure and thereby maintain the appropriate levels of sodium. The reactor vessel joint should not be exposed to full liquid sodium.

FIGURE 4.1-1

CRBRP HTS MATERIALS AND STRUCTURES

X. REACTOR VESSEL TRANSITION JOINT*

- PURPOSE
  - To describe the design and analysis of the low temperature reactor vessel transition joint
- SCOPE
  - Location, geometry, material selections, and fabrication
  - Environmental conditions
  - Structural evaluation
- CONCLUSIONS
  - The engineered design, service conditions and analysis establishes the integrity of the RV transition joint

*Responds to questions 210.3 and 250.5 related to PSAR 5.2.6

FIGURE 4.1-2

REACTOR VESSEL INTERNALS

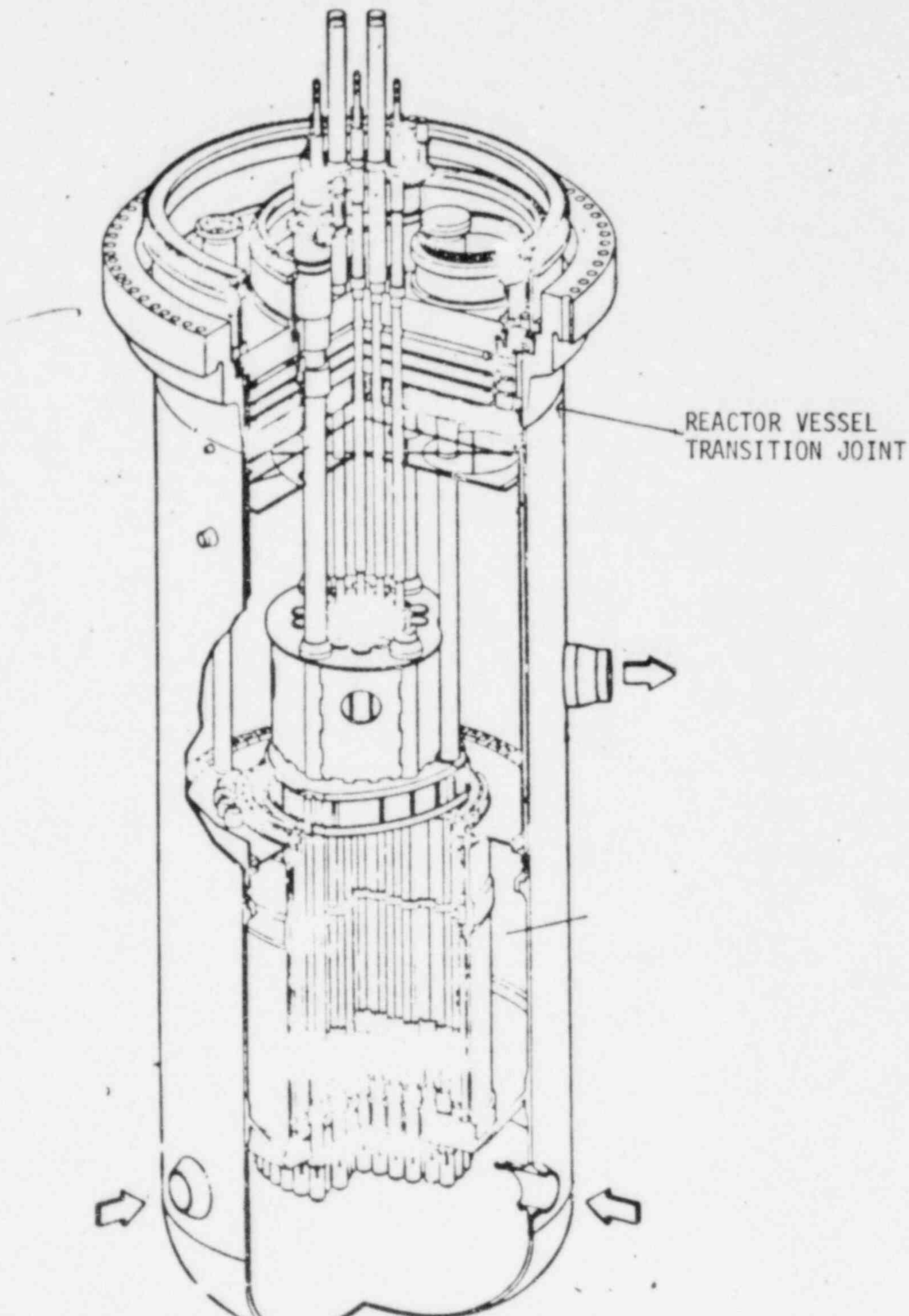


FIGURE 4.1-3

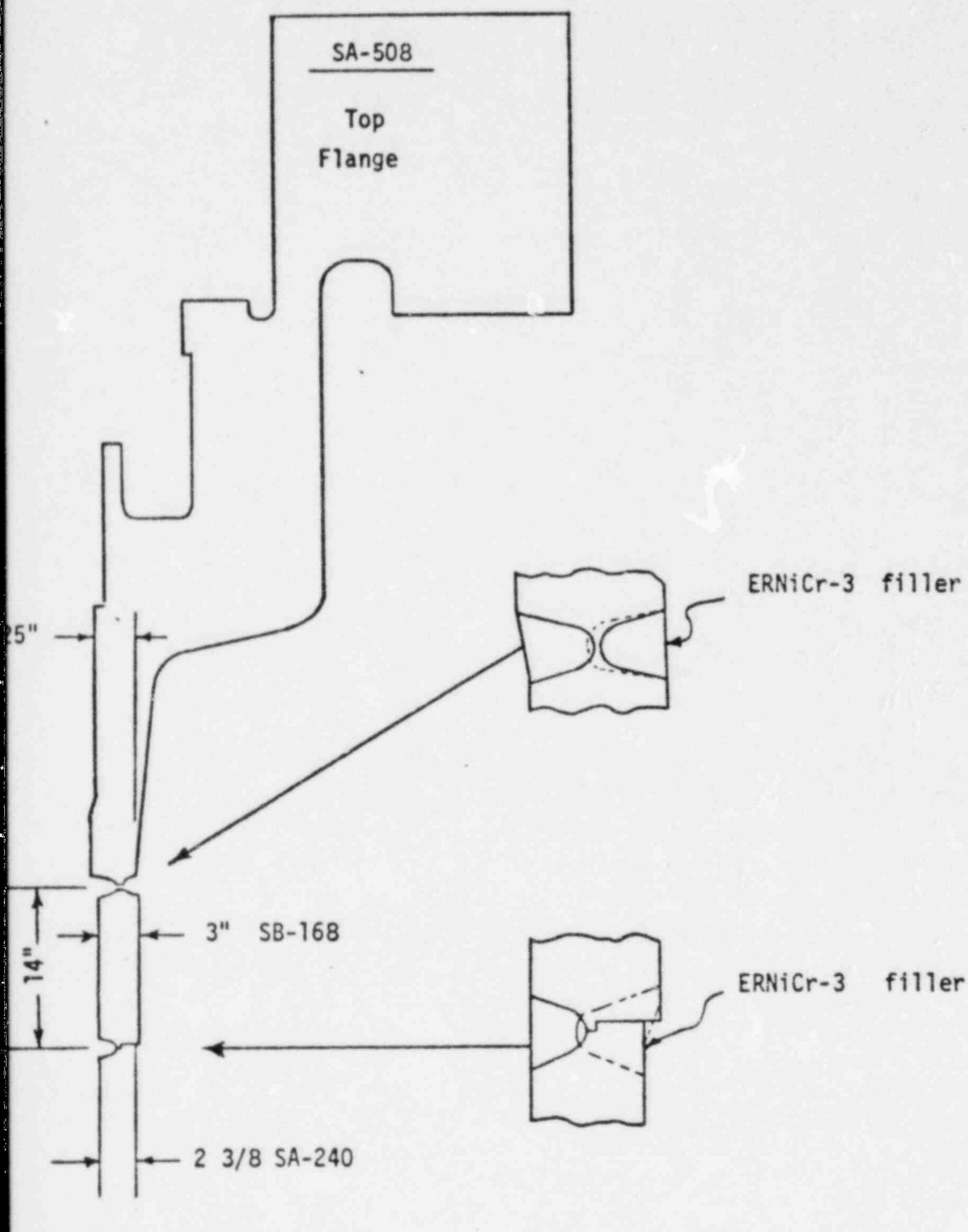

SA-508
Top
Flange
ERNiCr-3 filler
25"
3" SB-168
14"
ERNiCr-3 filler
2 3/8 SA-240

FIGURE 4.1-4 STANDARDS UTILIZED IN DISSIMILAR METAL VESSEL JOINT WELD

- Design, materials and fabrication in accordance with 1974 edition ASME Code (with Winter Addendum) and specified code cases

- In all cases, materials used are in accordance with ASME Code and supplemented by RDT standards (previous to NE standards)

SA 508 Class 2	-	M2-7T
SB 168 Type 1 annealed	-	M5-4T
SA 240 Type 304	-	M5-1T
SFA 5.14 Type ERNiCr 3	-	M1-11T

- Fabrication was in accordance with ASME Code and supplemented by RDT standards (previous to NE standards)

E 15-2NB-T	-	Class 1 Nuclear Components
F3-6T	-	Nondestructive examination
F6-5T	-	Welding and brazing qualifications

SA-508 TO SB-168 UPPER WELD

SEQUENCE OF OPERATIONS

1) FIT UPPER FLANGE TO INCONEL SHELL
2) WELD ID USING SUB-ARC PROCESS
3) BACKGROOVE OD USING ARC-AIR PROCESS
4) GRIND BACKGROOVE FOR PT
5) PT BACKGROOVE
6) WELD OD USING SUB-ARC PROCESS
7) PRELIMINARY STRESS RELIEVE (15 MIN. AT 1125°F)
8) GRIND ID & OD WELD SURFACES
9) PRELIMINARY PT OF ID & OD
10) PRELIMINARY RT (ONE NORMAL & ONE ANGLE SHOT USING WIDE SPECTRUM X-RAYS)
11) FINAL STRESS RELIEF AT 1125°F (10 HOURS)
12) MACHINE PREP FOR LOWER WELD
13) FINAL PT UPPER WELD
14) FINAL RT (ONE NORMAL SHOT USING WIDE SPECTRUM X-RAYS)

FIGURE 4.1-5

SB 168 TO SA 240 LOWER END
SEQUENCE OF OPERATIONS

- Fit upper vessel assembly to 304 SS shell. Use ship-lap joint to maintain concentricity.
- Weld I.D. using sub-arc process.
- Machine backgroove.
- PT backgroove.
- Weld O.D. using sub-arc process.
- Grind I.D. and O.D. for NDE.
- PT I.D. and O.D.
- Final RT using Iridium 192 isotopic source.

FIGURE 4.1-6

DISSIMILAR METAL REACTOR VESSEL WELD CONSIDERATIONS

- Review of transition joint technology and performance
- Location, design and materials selection for ferritic to austenitic transition joint aimed at mitigating failure mechanisms and environmental conditions
- Classification of factors that contribute to dissimilar weld joint failures
  - Thermal environment and cyclic loading
  - Low oxidation resistance of ferritic material (especially in region of fusion zone)
  - Carbon migration from the ferritic material
  - Metallurgical instability and deterioration caused by elevated temperature exposure

FIGURE 4.1-7

THERMAL ENVIRONMENT	-	TEMPERATURE LOW ~ 450°F JOINT LOCATED TO MINIMIZE MAGNITUDE OF THERMAL CYCLE DURING OPERATION
OXIDATION	-	O.D. ENVIRONMENT - NITROGEN I.D. ENVIRONMENT - ARGON, NA VAPOR OR FILM
CARBON MIGRATION	-	NI BASE FILLER METAL TO MINIMIZE MIGRATION DURING WELDING AND POST WELD HEAT TREATMENT DIFFUSION COEFF. ~ 10^{-16} CM^2/SEC (450°F) ~ 10^{-9} CM^2/SEC (1100°F)
METALLURGICAL STABILITY	-	PHASES PRESENT AFTER POST WELD HEAT TREATMENT SHOULD NOT UNDER GO CHANGE $M_{23}C_6$ NOT EXPECTED TO FORM

FIGURE 4.1-8

FIGURE 4.1-9 OTHER FACTORS

IRRADIATION

- Low total fluence $<10^{17}$ n/cm^2 (E > 1 Mev) at end of life and is below threshold for mechanical property degradation of either the ferritic or austenitic materials. Surveillance not required per 10CFR50, Appendix H.

DECARBURIZATION OR CORROSION DUE TO SODIUM FILM

- Temperature is ~450°F and both sodium corrosion and interstitial transfer are undetectable.

NITRIDING OF FERRITIC MATERIAL

- Gas phase nitriding will not occur below 600°F, thus external vessel environment will not degrade material.

OTHER CORROSION MECHANISMS

- Fabrication sequence prevents furnace sensitization of stainless steel — precludes intergranular attack during storage and erection.

- Galvanic and other types of corrosion precluded during operation by O.D. and I.D. environments.

FIGURE 4.1-10

SERVICE EXPERIENCE IN TEMPERATURE
RANGE BELOW 600°F

- PWR SAFE END WELD PERFORMANCE SERVICE TEMPERATURE
  ~550°F MATERIALS SA508 - INCONEL 82 - TYPE 304SS
  NO SERVICE FAILURES EXPERIENCED

CONCLUSIONS

- Dissimilar metal reactor vessel materials will provide safe, reliable performance

- Material degradation is not anticipated due to low service temperature coupled with benign environment

- Experience with this material combination in PWR service confirms material adequacy for this service temperature

FIGURE 4.1-11

REACTOR VESSEL TRANSITION JOINT STRESS ANALYSIS - PART 2

This presentation will continue what Len France discussed about the reactor vessel transition joint and describe the stress analysis in response to questions 210.3 and 250.5 (Figure 1).

Figure 2 discusses the loading conditions, general environment, and other things about the joint. As Len mentioned, the joint is a low-temperature joint; therefore, it was analyzed in accordance with Section NB of the ASME Code. He also noted that the joint is always above the sodium level, and the steady-state stresses are due entirely to axial gradients that exist at the weld. Transient stresses at this elevation are due to changes in the axial gradient. There is no significant radial temperature gradient at any time because there is no sodium at the weld. The transients are very slow in this area because this is above the sodium level. The analysis was performed with reactor transients that were conservatively grouped and because it is out of the sodium, the joint sees none of the high cycle thermal striping-type conditions associated with a fluid to metal interface.

The only significant primary stresses in this joint are due to seismic. It was mentioned that irradiation effects are negligible. There is a very small nominal cover gas pressure and completely negligible pressure fluctuations exist at the joint because the joints are in the cover gas.

The joints were analyzed with several finite element models, utilizing the ABSA program and constant strain elements. The coarse mesh model is shown in Figure 3. The right hand part of this Figure is attached at the bottom of the left half and extends well below the joint. The lower joint is approximately at elevation -63, and the upper joint is approximately at elevation -49. The sodium level is approximately at elevation -87, well below the weld. The model extends well above and below the joints to eliminate any local effects brought in by locally applied loads and boundary conditions.

The analysis was elastic in accordance with Section III NB rules. Material properties used in the analysis were from ASME Section III. The materials properties change at the weld center line, as shown in Figure 4.

The loads and boundary conditions applied to the vessel are shown in Figure 5. The vessel is supported by the reactor vessel support, with a pre-loaded bolt going through the flange, support, and into the ledge. The closure head is supported from the vessel. There is a slight external pressure from the insulation straps that hold insulation on the upper part of the vessel. The internal cover gas pressure starts from the head seal and extends down past the two welds to the sodium level. At that point there is an increase in pressure due to the sodium head. The model is also loaded with the weight of the reactor vessel.

The sodium that contacts the reactor vessel at this location is the sodium that comes from behind the reactor vessel liner. It is not the 1,000-degree output plenum sodium, but it's the 800-degree sodium that comes in at the inlet plenum and is gradually heated up from behind the liner to 855°F at the top of the pool. The lower weld is at the temperature of 615 Fahrenheit. The upper weld is at 443 Fahrenheit. There is a linear gradient in between the temperatures shown on Figure 6.

And again, because of the insulation on the outside of the guard vessel, and the fact that there is no sodium at the weld, very little, if any, radial gradient exists at these weld locations.

A series of refined mesh models were developed with axisymmetric constant strain elements to confirm the acceptability of the previously defined model. Both models were analyzed for the steady state loading conditions. The steady state results of the previously defined model were used as loads on the refined model. These results showed that the previous model provided acceptable results at the upper weld, but did not provide acceptable results at the lower weld for peak stress intensity effects. As a result of this study, the peak stress values from the previous model, at the lower weld, were increased by a ratio of the results from the refined model to the previously defined model. The first refinement model of both welds is presented in Figures 7 through 9. The second and third refinements of the lower weld are presented in Figures 10 and 11, respectively.

The location of stress classification lines are shown in Figure 12. Stress lines 12 and 13 are at the lower weld center line. Seventeen and eighteen are at the upper weld center line.

I mentioned before that the steady-state primary stresses are negligible in both joints. They are on the order of 1 ksi membrane stress and 1-1/2 ksi bending plus membrane stress as shown in Figure 13. These stresses are due to the pressure and the dead weight. The seismic loads (for the OBE) have a more significant effect on these joints. OBE seismic stresses are maximum on the order of 17.3 ksi membrane stress as shown in Figure 14. There is no bending stress in the shell from the seismic loading condition, but there is a slight peak effect due to the discontinuities in the joints where there are some changes in thickness. The primary stress limits are applied to the mechanical seismic loading plus the loadings from the dead-weight and pressure.

The minimum margin of safety for this condition occurs in the 304 shell where there is a 1 ksi pressure plus dead-weight membrane stress and the 15 ksi stress due to the earthquake, which are combined to give a 16.24 ksi load. The allowable at that location is 16.3 ksi. So that is within the allowable which does not vary significantly with increasing temperature.

Thermal transients produce the major shell stress, and these are quoted as Primary plus Secondary stress intensity ranges, which vary from 35 ksi near the upper weld to as high as 59 ksi in the 304 stainless steel shell as shown in Figure 15. The Primary plus Secondary plus Peak stress intensity ranges shown in this figure are due to peak stress amplification factors associated with the refined models used for this analysis. The models are developed with four node constant strain elements; there are six elements through the thickness in the coarse model, 12 for the refined upper weld model and 36 in the refined lower weld model.

With regard to transients, there is at least one, the U2B uncontrolled rod withdrawal, which does produce an increase in temperature for a five minute period.

The other transients are all defined in the specification. The transients are conservatively grouped to get the maximum up and down transients, and then the analysis is based on that group. The total number of cycles are added up. Therefore, the total number of events is the actual number of events, but the classification of them produces more severe events within each group. The peak stress is primarily due to the non-linear sheer stress distribution in the welds, the shear stress distribution through the thickness.

Figure 16 shows the upper weld primary plus secondary stress intensity ranges which are all within the allowable limits.

There are no secondary seismic stresses. This is the primary weight, the seismic, and the thermal, and in this combination, the seismic amplitudes that were quoted before were multiplied by two to get the total seismic range and added to the thermal stress intensity range. Doubling the seismic, and adding it to the range from the thermal, implies that the seismic event is occurring at both ends of the thermal transient.

Now there is some shear in that primary plus secondary stress intensity number. We combined the shear by using the ASME stress component methods. The average shear stress is used in the P_m and P_L stress intensity categories. The peak stress difference between the initial and refined models is significant. The peak stress data shown on Figure 16 includes an increase in peak stress associated with the model refinement.

This explains the fact that the numbers in the primary plus secondary columns are so much lower than the numbers with the peak stresses included and we use a very small concentration factor only where the thickness changes a little bit. The full details of the approach used are included in ASME paper #75-PVP-63.

Regarding documentation, we've submitted documents that are consistent with the typical licensing rules which are to give the criteria in the approach to the structural evaluation at the PSAR stage and get into the details with the completed analysis at the FSAR. Information on how the stress components were combined is included in the previously mentioned paragraph.

There are no anchor point motion type stresses that might come in from the piping into this area. The piping would have a completely negligible effect on this area. The expansion stress as you heat the vessel up from 70 degrees would be included.

Now as Figure 17 shows, at the lower joint, the NB3228.3 analysis was performed on the Inconel 600 shell and the results of that analysis showed that all the criteria of that paragraph were met. The P_L plus P_B plus Q range, excluding thermal bending stress, is less than 3 S_m. The fatigue analysis results are presented here at the critical location. In the lower joint, there is a fatigue usage factor of 0.162 with an allowable of 1.0, using the simplified elastic plastic rules. The thermal ratchetting requirements of NB3222.5 are met. The temperature and materials are within the specific required limits. This shows that the joint is clearly adequate. As has been pointed out, this was all done to the design rules and procedures of Section III.

The lower part of the reactor vessel, below the operating sodium level, is over the 800-degree limit and a more refined elastic plastic model of the lower shell was developed to analyze that area near the top of the sodium elevation.

The fatigue usage factor was 0.25 at the sodium surface elevation. That elevation had higher stresses and higher temperature than the part of the 304 shell where the 3 S_m limit is exceeded and, therefore, the lower 304 shell at that elevation will have a fatigue usage factor of less than 0.25, including the plasticity figures. This does not really see the outlet plenum sodium temperature and I doubt that it gets very much above it.

As previously mentioned, the normal operating temperature at the lower weld is 855°F. This temperature may rise slightly for very short periods of time because one of the outlet plenum transients exceeds the normal operating temperatures for a period of five minutes for each of sixteen cycles. There are about 16 of those transients. So, the time at temperatures above normal operating temperatures would be very short.

To sum up the analysis (Figure 18), the joint is a low temperature joint which was analyzed in accordance with NB Section III. All required criteria were met: primary membrane, primary plus secondary membrane plus bending and the fatigue criteria are met. Finally, the design service conditions and analyses establish the integrity of this joint.

REACTOR VESSEL TRANSITION JOINT
STRESS ANALYSIS

G. H. NICKODEMUS, MANAGER
STRESS ANALYSIS

WESTINGHOUSE ADVANCED REACTORS DIVISION

THIS PRESENTATION IS PART OF THE
RESPONSE TO QUESTIONS 210.3 AND 250.5

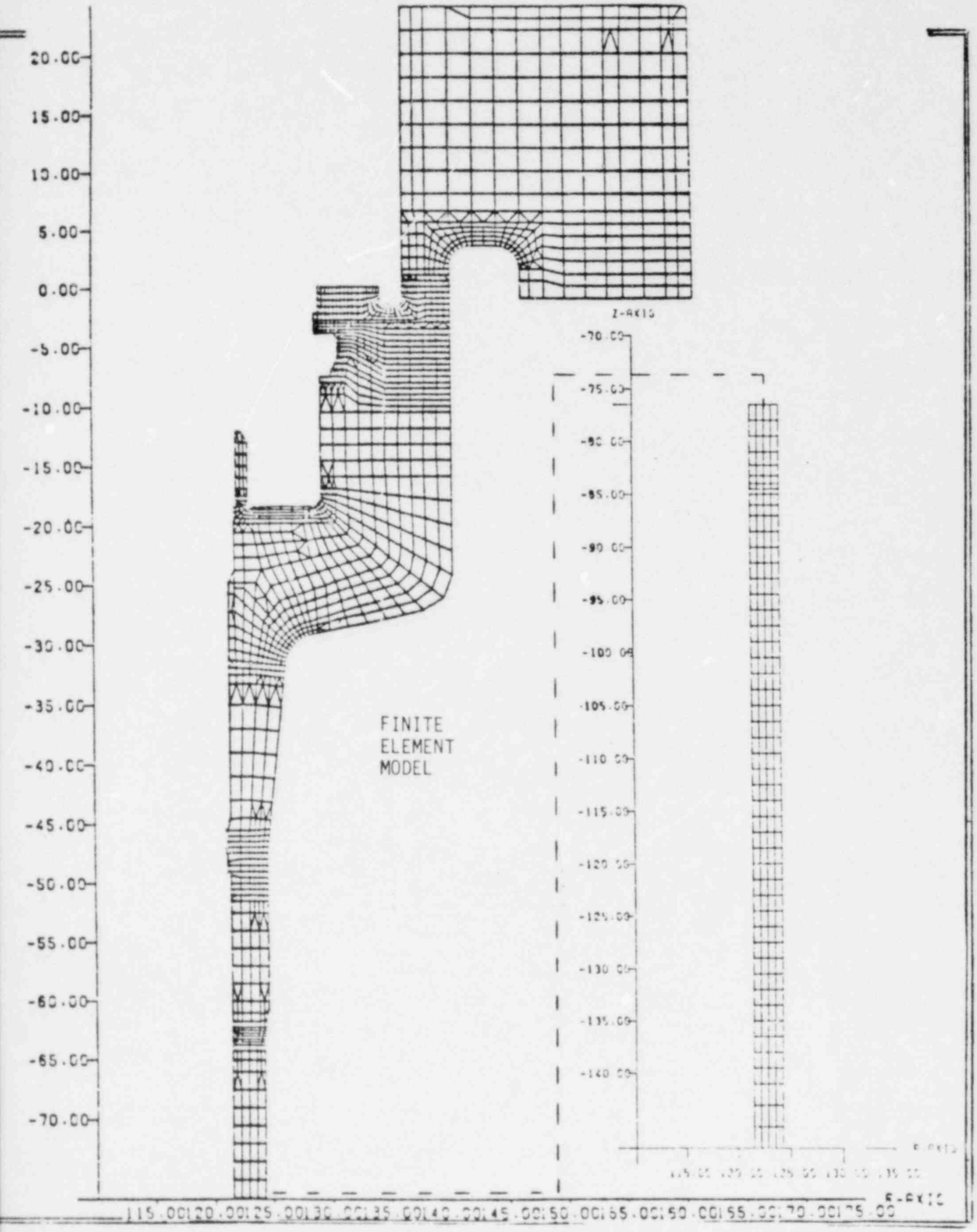
FIGURE 4.2-1

REACTOR TRANSITION WELD
LOADING CONDITIONS

- STEADY STATE AND TRANSIENT TEMPERATURES ARE ALL WITHIN THE APPLICABLE RANGE OF THE ASME SECTION III CRITERIA.
- STEADY STATE STRESSES ALMOST ENTIRELY DUE TO AXIAL TEMPERATURE GRADIENT.
- TRANSIENT STRESSES ARE DUE TO CHANGES IN THE AXIAL GRADIENT.
- NO SIGNIFICANT RADIAL GRADIENTS EXIST AT ANY TIME.
- TRANSIENTS ARE VERY SLOW BECAUSE THE OUTLET PLENUM SODIUM DOES NOT CONTACT THE REACTOR VESSEL, AND THE WELDS ARE WELL ABOVE THE SODIUM LEVEL.
- REACTOR TRANSIENTS ARE CONSERVATIVELY GROUPED.
- THE ONLY SIGNIFICANT PRIMARY STRESSES ARE DUE TO SEISMIC EVENTS.
- IRRADIATION EFFECTS ARE NEGLIGIBLE.
- LOW NOMINAL PRESSURE AND NEGLIGIBLE PRESSURE FLUCTUATIONS.
- NO HIGH CYCLE THERMAL FLUCTUATIONS BECAUSE THE WELDS ARE ALWAYS ABOVE THE SODIUM LEVEL.

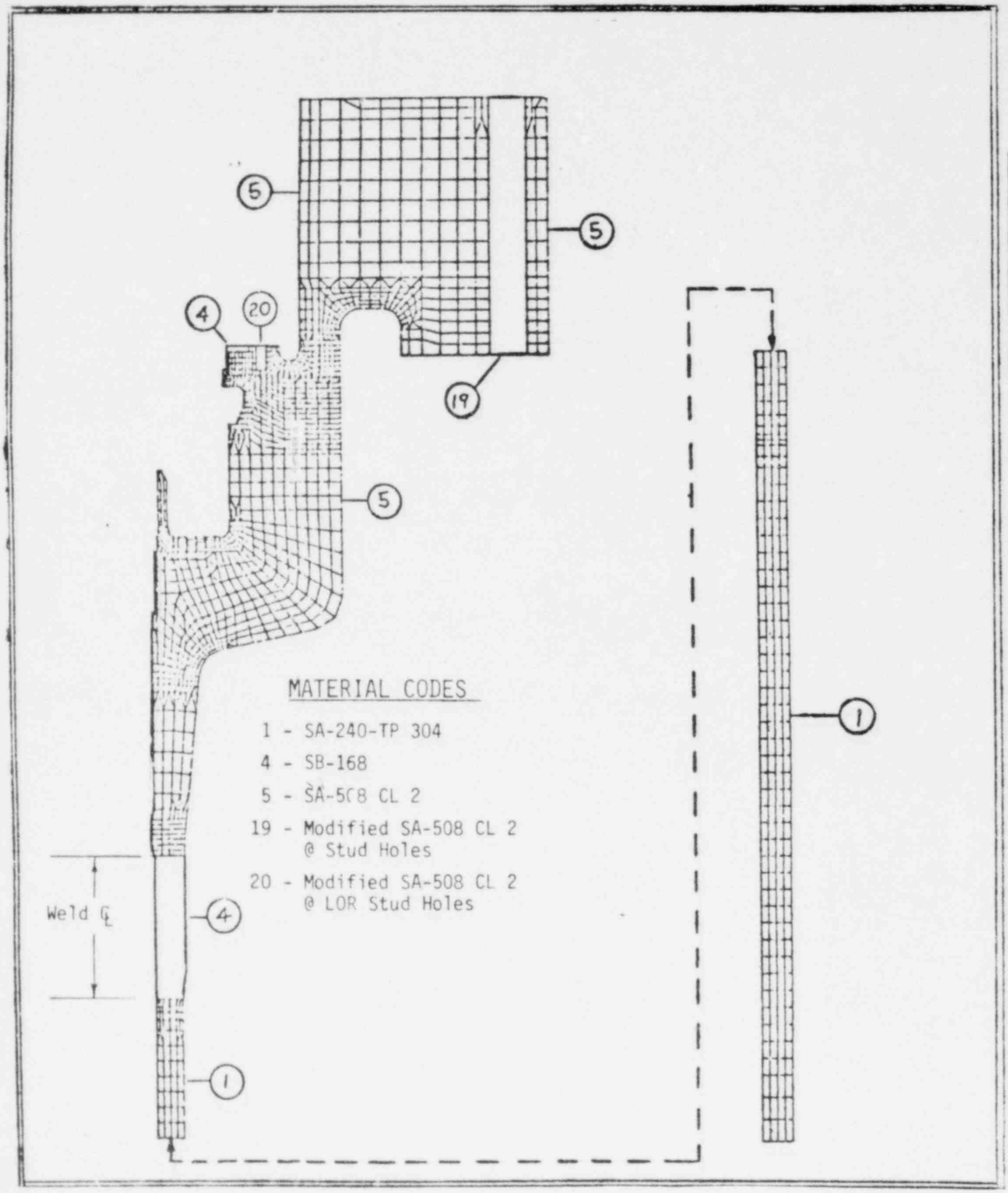
FIGURE 4.2-2

FIGURE 4.2-3



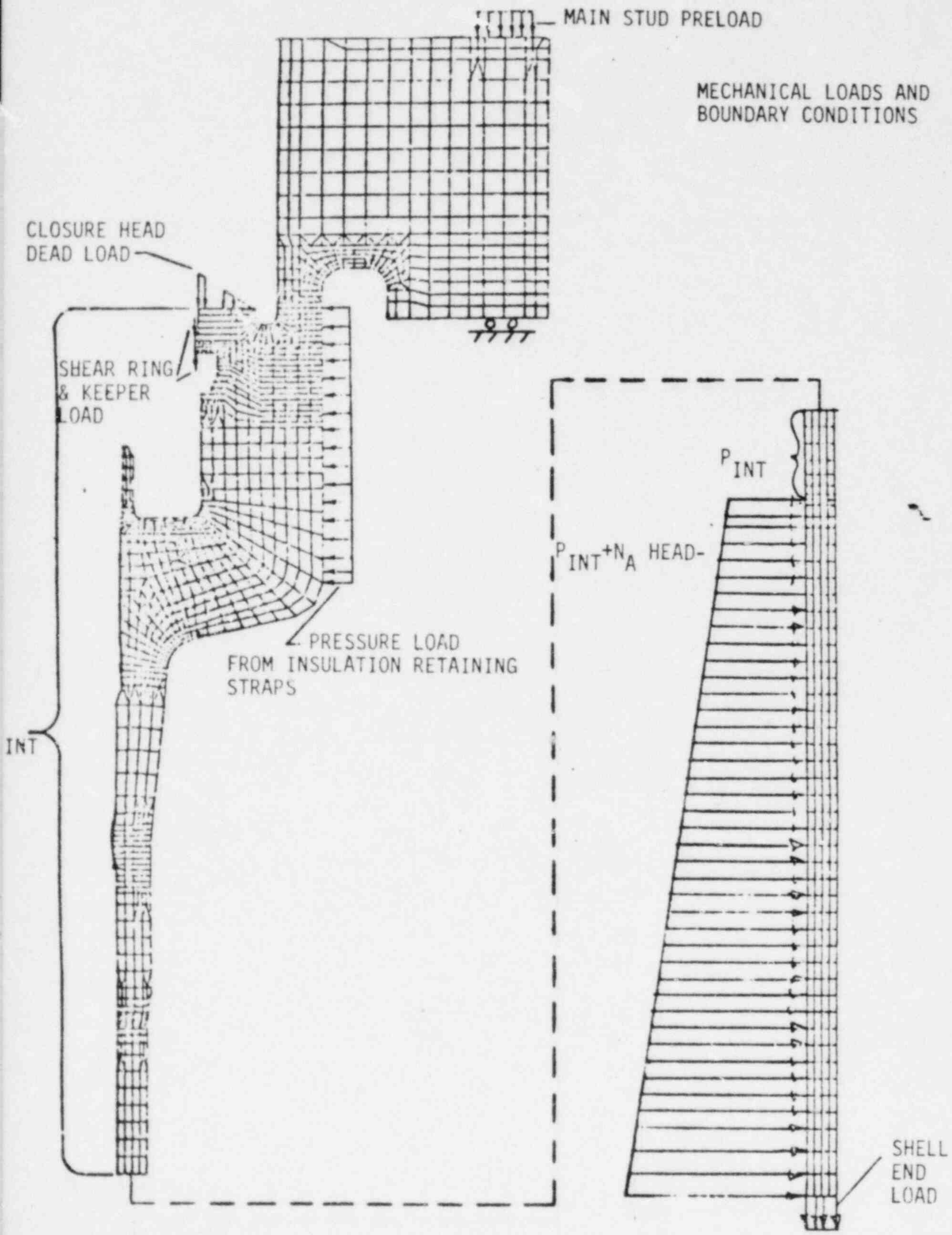
4

FIGURE 4.2-4



5
5
4
20
19
5
MATERIAL CODES
1 - SA-240-TP 304
4 - SB-168
5 - SA-508 CL 2
19 - Modified SA-508 CL 2 @ Stud Holes
20 - Modified SA-508 CL 2 @ LOR Stud Holes
1
Weld ℄
4
1

FIGURE 4.2-5



MAIN STUD PRELOAD
MECHANICAL LOADS AND
BOUNDARY CONDITIONS
CLOSURE HEAD
DEAD LOAD
SHEAR RING
& KEEPER
LOAD
P_{INT}
$P_{INT}+N_A$ HEAD
PRESSURE LOAD
FROM INSULATION RETAINING
STRAPS
INT
SHELL
END
LOAD

STEADY STATE AXIAL
TEMPERATURE GRADIENT

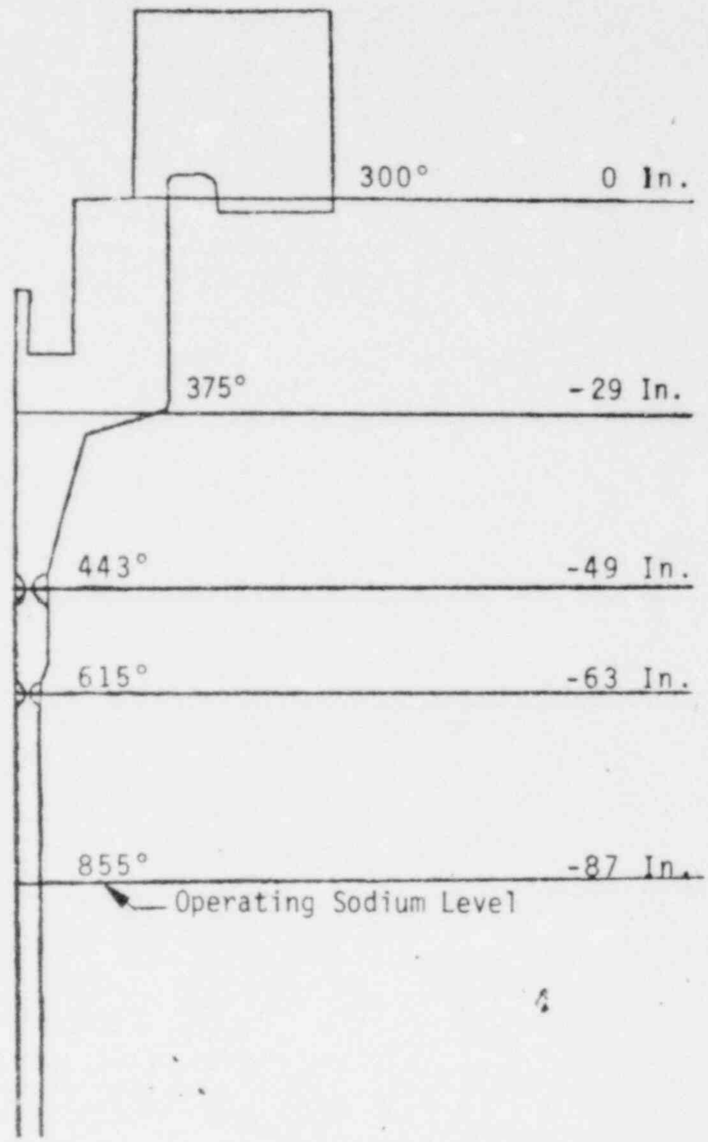
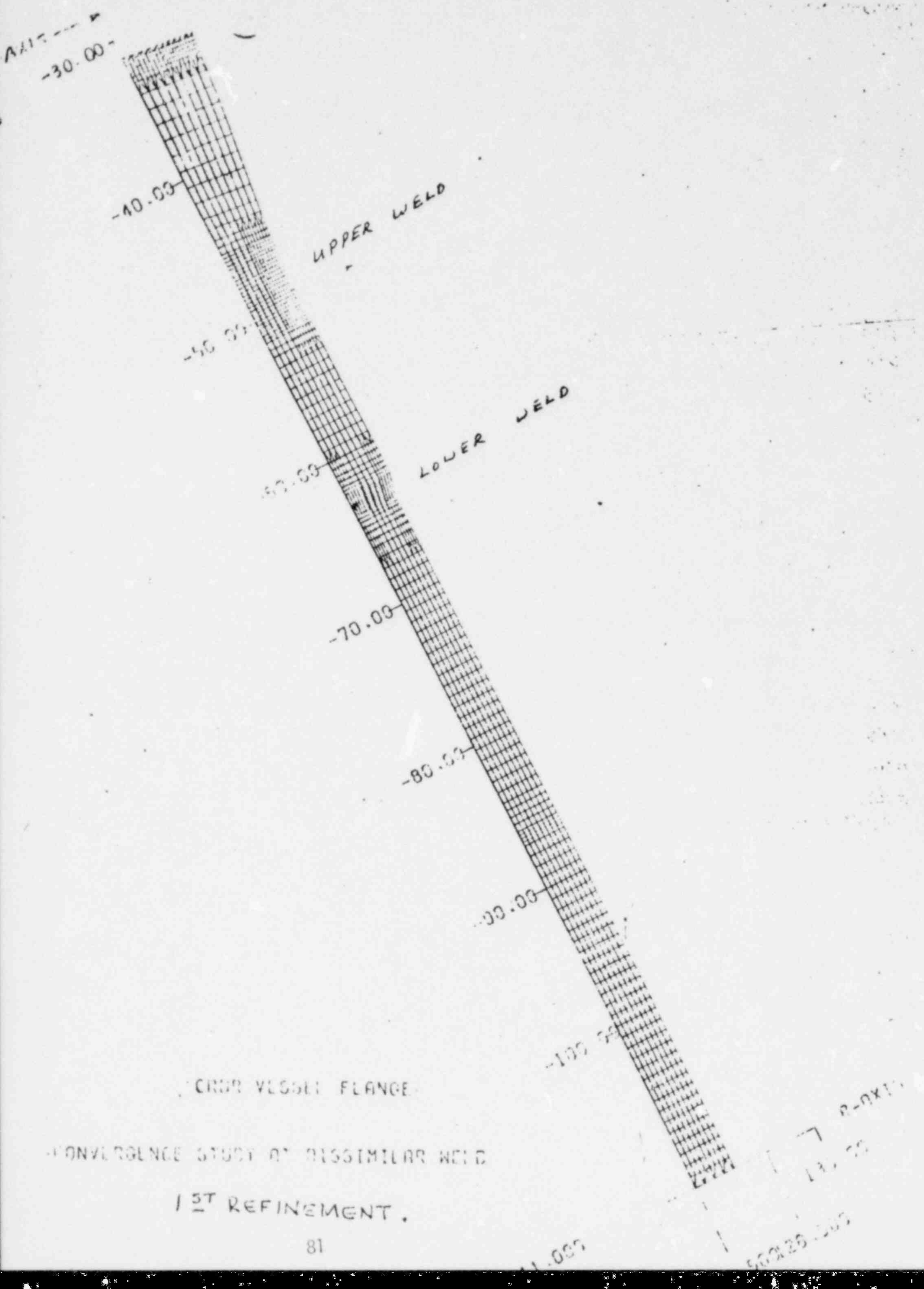


FIGURE 4.2-6

FIGURE 4.2-7

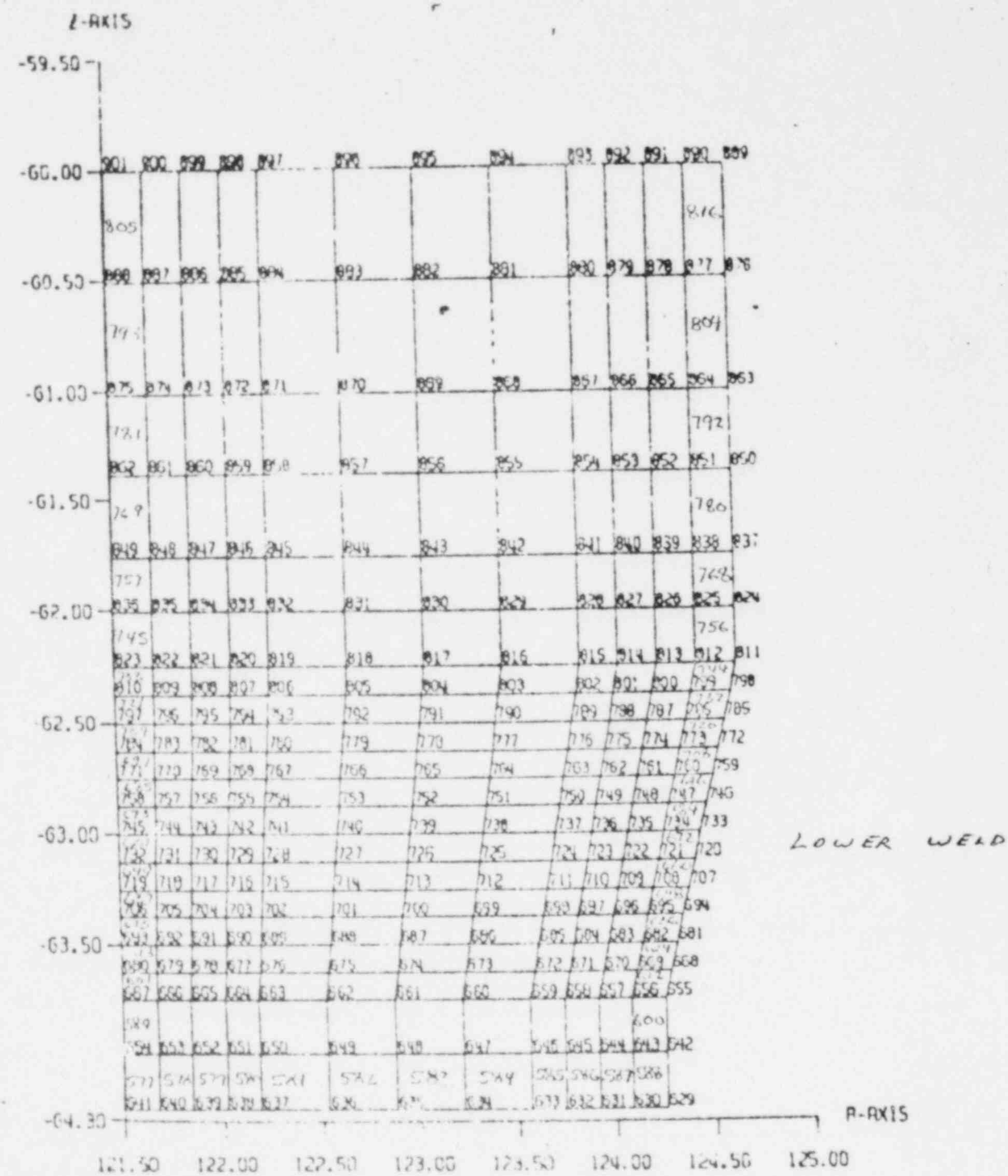


CRDR VESSEL FLANGE

CONVERGENCE STUDY AT DISSIMILAR WELD

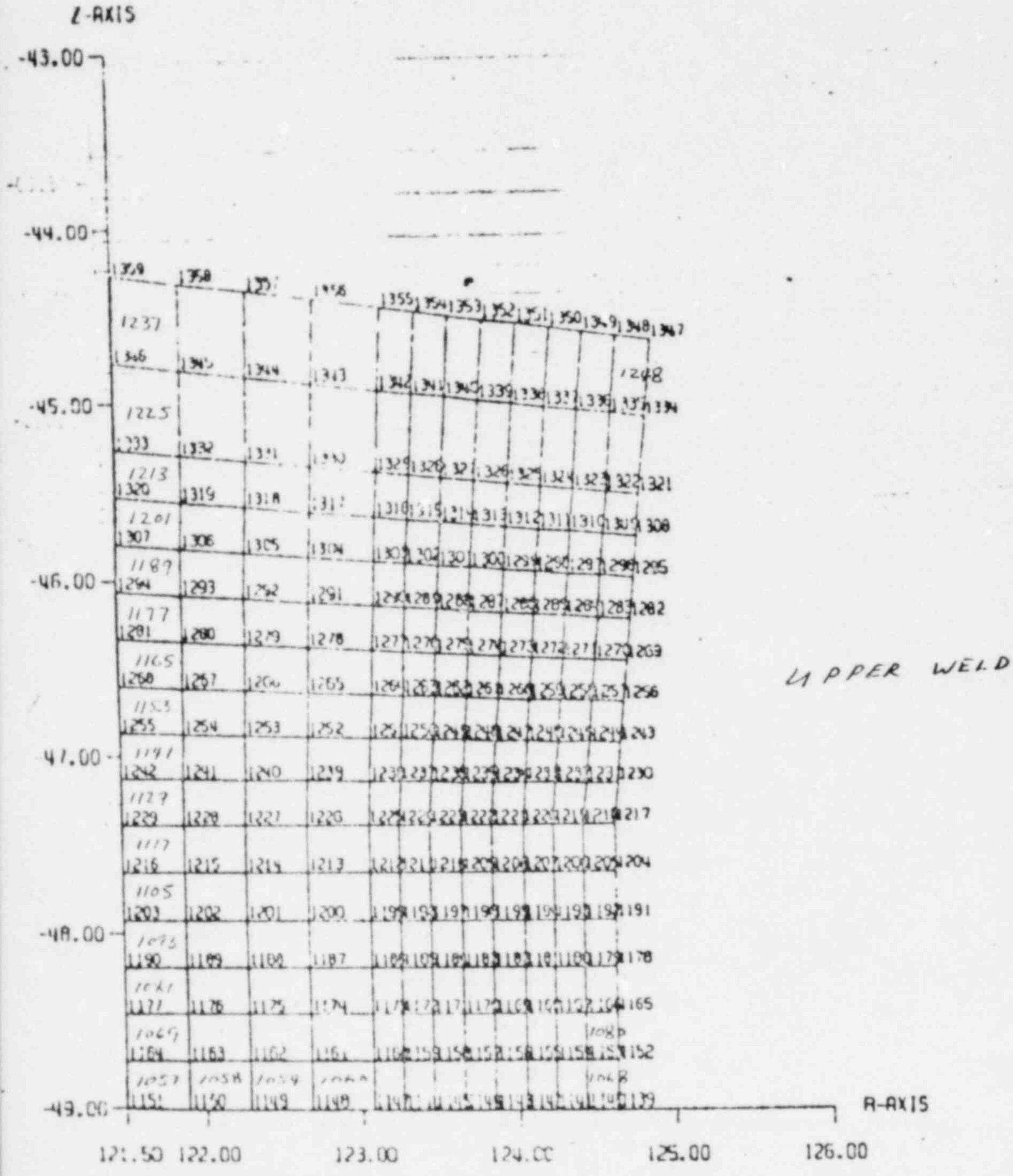
1ST REFINEMENT.

FIGURE 4.2-8



CRBR VESSEL FLANGE CONVERGENCE STUDY AT DISSIMILAR WELD
DETAIL VIEW NO. 8. FIRST REFINEMENT

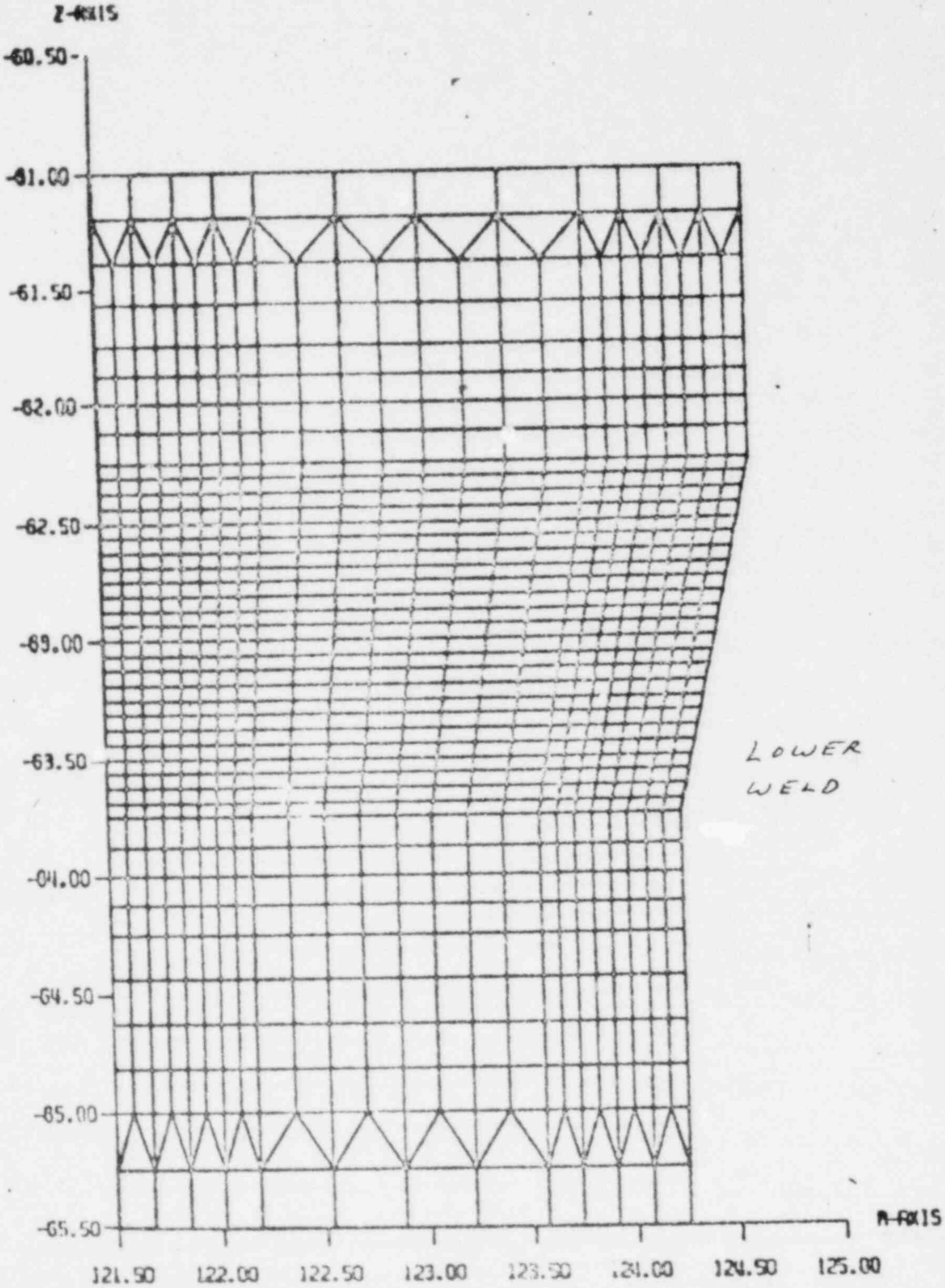
FIGURE 4.2-9



CABA VESSEL FLANGE-CONVERGENCE STUDY AT DISSIMILAR WELD

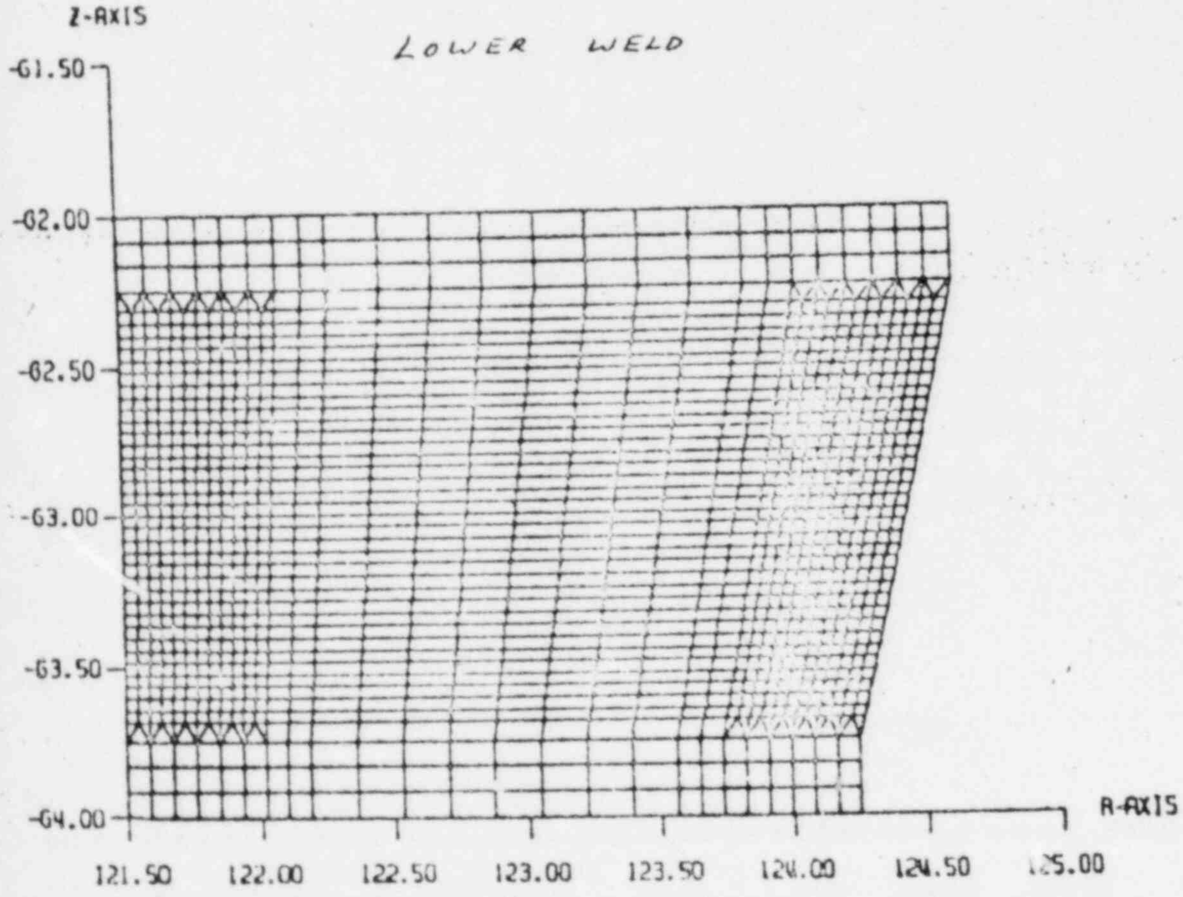
DETAIL VIEW NO. 11. FIRST REFINEMENT

FIGURE 4.2-10



CIC VESSEL FLANGE CONVERGENCE STUDY AT DISSIMILAR WELD-2ND REFINEMENT

FIGURE 4.2-11



CIC VESSEL FLANGE CONVERGENCE STUDY AT DISSIMILAR WELD-3RD REFINEMENT

FIGURE 4.2-12

STRESS CLASSIFICATION LINES

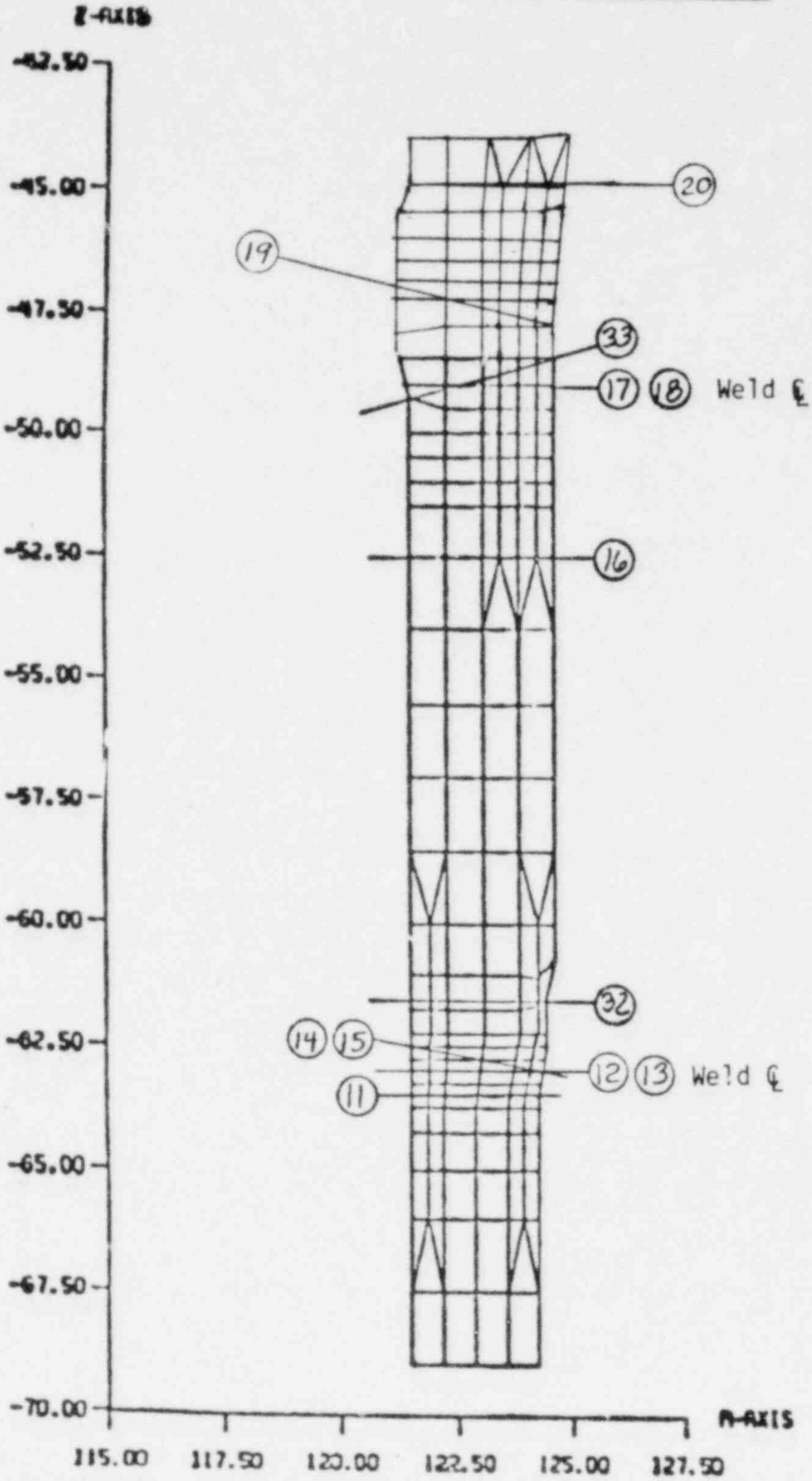


FIGURE 4.2-13

STEADY STATE PRIMARY STRESSES

100% POWER

$$\left.\begin{array}{l} P_m = 1.0 \text{ ksi} \\ P_1 + P_b = 1.5 \text{ ksi} \end{array}\right\} \text{at both welds}$$

9

FIGURE 4.2-14

SEISMIC ANALYSIS

STRESS INTENSITY AMPLITUDES

WELD MATERIAL	SCL	PRIMARY MEMBRANE[1]	PRIMARY + SECONDARY + PEAK[2]
UPPER WELD			
SA-508	33	17.32 KSI	17.32 KSI
	19	17.32	20.79
	18	17.32	17.32
INCONEL 600	17	17.32	17.32
	16	15.24	15.24
LOWER WELD			
INCONEL 600	32	15.24	19.93
	15	15.24	15.24
	14	15.24	15.24
	13	15.24	15.24
SA-240 TP304	12	15.24	15.24
	11	13.12	15.15

MINIMUM MARGIN OF SAFETY FOR COMBINED PRIMARY MEMBRANE SI IS IN THE 304 SS WHERE THE COMBINED SI IS < 1.5 + 13.12 OR < 14.52 KSI.

1. ALLOWABLE IS S_M, WHERE S_M IS 23.3 KSI FOR INCONEL 600, 26.7 KSI FOR SA 508, AND 16.3 KSI FOR 304 SS.

2. PEAK STRESS INTENSITY IS EQUAL TO THE PRIMARY MEMBRANE TIMES A LOCAL STRESS CONCENTRATION FACTOR.

FIGURE 4.2-15

THERMAL TRANSIENT ANALYSIS

4 UP TRANSIENTS AND
5 DOWN TRANSIENTS ANALYZED

RESULTS: STRESS INTENSITY RANGES

MATERIAL	SCL	PRIMARY & SECONDARY	PRIMARY & SECONDARY + PEAK
UPPER WELD			
SA-508	33	39.6 KSI	90.3 KSI
	19	34.8	40.6
	18	42.8	58.7
INCONEL 600	17	45.5	63.3
	16	48.5	48.5
LOWER WELD			
INCONEL 600	32	39.9	45.7
	15	51.5	81.6
	14	53.4	81.2
	13	47.4	84.3
SA-240 TP304	12	45.1	73.3
	11	58.9	63.1

FIGURE 4.2-16

THERMAL & SEISMIC ANALYSIS

To get Seismic Stress Intensity Ranges, the Seismic Stresses on previous tables are multiplied by 2.

Material	SCL	Primary & Secondary	$3\ S_M$ Allowable	Primary & Secondary + Peak	Usage Factor
Upper Weld					
SA-508	33	57.7 ksi	80.1 ksi	105.0 ksi	<.162
	19	62.7	80.1	76.6	
	18	70.7	80.1	72.7	
Inconel 600	17	64.9	69.9	92.8	
	16	61.9	69.9	70.1	
Lower Weld					
Inconel 600	32	70.4	69.9	79.3	<.162
	15	75.6		105.7	<.162
	14	77.5		105.3	.162
	13	68.4		105.3	.069
SA-240 TP304	12	66.1	49.1	83.8	<.25
	11	77.9	48.9	76.3	<.25

FIGURE 4.2-17

THERMAL & SEISMIC STRESS ANALYSIS

- THE UPPER WELD PRIMARY PLUS SECONDARY STRESS INTENSITY RANGE IS WITHIN THE ALLOWABLE.
- THE LOWER WELD PRIMARY PLUS SECONDARY STRESS INTENSITY RANGE EXCEEDS THE 3 S_M ALLOWABLE. THEREFORE:
  - ADDITIONAL ANALYSES PER NB 3228.3 WERE PERFORMED FOR THE INCONEL 600 SHELL.
  - ALL THE REQUIREMENTS OF THIS PARAGRAPH ARE MET
    - $P_L + P_B + Q$ RANGE, EXCLUDING THERMAL BENDING STRESS $\leq 3\ S_M$
    - THE RESULTS OF THE FATIGUE EVALUATIONS ARE PRESENTED ABOVE
    - THE THERMAL RATCHETTING REQUIREMENTS OF NB-3222.5 ARE MET
    - THE TEMPERATURES AND MATERIAL PROPERTIES ARE WITHIN THEIR RESPECTIVE LIMITS.
  - AN ELASTIC PLASTIC ANALYSIS OF THE REGION BELOW THE LOWER WELD RESULTED IN A USAGE FACTOR OF 0.25. THE USAGE FACTOR IN THE 304 SS SHELL AT THE LOWER WELD IS < 0.25 DUE TO LOWER TEMPERATURES AND STRESSES.

FIGURE 4.2-18

CONCLUSIONS

- THE REACTOR VESSEL TRANSITION JOINTS ARE LOW TEMPERATURE REGIONS ANALYZED IN ACCORDANCE WITH THE ASME SECTION III CRITERIA.

- ALL REQUIREMENTS OF THE CRITERIA ARE MET

  - PRIMARY MEMBRANE STRESS INTENSITY CRITERIA ARE MET

  - PRIMARY PLUS SECONDARY MEMBRANE PLUS BENDING STRESS INTENSITY CRITERIA ARE MET

  - FATIGUE CRITERIA ARE MET

- THE ENGINEERED DESIGN, SERVICE CONDITIONS, AND ANALYSIS ESTABLISH THE INTEGRITY OF THE REACTOR VESSEL TRANSITION JOINTS.

IHTS TRANSITION JOINT

By

A. Dalcher (GE)

5566B-444B:2
(S3597) 1

IHTS TRANSITION JOINT

You already have some familiarity with the transition joints that are in the intermediate heat transport system, but in this discussion (Figure 1) I would like to review them with you.

The IHTS piping route shown in Figure 2 has each transition joint numbered. We have two transition joints in the superheater hot leg, Numbers 1 and 3, that operate at temperatures normally above 700° Fahrenheit. All the other joints that you see here are cold leg joints. They normally operate at temperatures below the creep range for their materials of construction.

First I will talk about our design evaluations (Figure 3). I will give you some background on the design criteria that we applied; the description of transition joints; the analysis methods; and information that supports our design evaluation that we have received from supporting programs. Our conclusions are summarized below and in Figure 4.

All the cold joints pass all the ASME Code criteria for the full 30-year life of the plant.

The hot joints pass all the ASME requirements for 15-year service. This doesn't mean that the joints will not survive beyond 15 years, but it does indicate that the analysis done to date shows satisfactory life or satisfactory conditions for 15-year operation.

The transition joint life tests, or the testing program results, indicated that our component integrity is certainly confirmed for times greater than the 15-year life.

The life tests also have shown that they confirmed the location of the most severely loaded or most severely exercised part of the transition joint as we had predicted from analysis. The life test program does confirm our analytical predictions.

5566B-444B:2
(S3597) 2

I will now discuss the design criteria briefly (Figure 5). Figure 6 shows the 1592-7 elevated temperature code case criteria, with strain and deformation limits from that code case. We use half of the permitted total plastic strain. Half of the total permitted by the code case is one-half, one, and two-and-a-half percent.

Creep fatigue damage usage factor less than 100 percent has been mentioned before, 1974 Code with 1975 summer addenda and the code case, and a set of RDT standards that assures us that high quality fabrication as well as quality assurance and structural analysis procedures are followed.

First let me present a description of the IHTS transition joints (Figure 7). The table in Figure 8 shows some detail about the 11 transition joints that were shown in Figure 2. The sizes vary from 26-inches for Joint Number 1 down to 3-inches for some of the vent lines.

There are just the two joints that are what we call hot leg joints. Those do turn out to be the most critical joints as you might expect.

TRANSITION JOINT FABRICATION

Now, Figure 9 is somewhat of a cartoon and is not to scale. The idea is just to show generally how these transition joints are made and put into a typical installation. It shows that there will be a shop fabricated spoolpiece of 2 1/4 Cr-1 Mo steel welded to alloy 800H welded to austenitic stainless steel. This would be the 316 or 304. It is possible that there could be an additional piece put on, depending on where the spoolpiece is located and exactly how they decide to construct the plant. If it is convenient to have another piece on there so they can make the field weld beyond that point, then it will be done that way.

But as we envision it now, there will be a field weld at each end. So, this finished piece is all shop-fabricated and is sent to the plant. Then it is installed in the field, with field welds to like-material on the straight section that comes from the vessel to which it is being attached at this end, and to the piping at the other end.

Figure 10 shows generally how a transition joint is fabricated. The pipe material is oversized both in its outside diameter and inside diameter. It is machined to a configuration that has been determined to be most advantageous from the standpoint of stresses that are left in this weld after it has been made. The root opening is also determined on a similar basis.

After these are fit up, the weld is deposited using the procedure which Mr. Patriarca has described. When that is all done, and also after it has been post-weld heat treated, there is machining done both on the outside and the inside, so that the root pass and the step are machined away. Similarly, any suck-in or whatever nonuniformities there might be on the OD are also machined away. So, we end up with a straight, smooth piece, smooth both inside and outside. There are no geometric discontinuities in this weld when it is done.

MATERIALS OF CONSTRUCTION

Information about the materials of construction are shown in Figure 11. Alloy 800H was chosen as the transition material. It is Code material with an intermediate coefficient of expansion. It is made long enough so that there is no stress carry-over from one joint to the next. That is, if you think about a characteristic length of a shell, the joints are far enough apart that one joint does not produce stress in the upstream or downstream joints.

The Inco 82 weld metal, again, is a Code material, with an intermediate temperature coefficient. It is a high-strength material and it does about the best job of any material to keep carbon migration from the 2 1/4 Cr-1 Mo toward the austenitic materials.

ER 16-8-2 weld metal (Figure 12) joins the 800H to the 304/316. This Code material has the mechanical properties of the kind that the engineers look for to get a good weld and this is the proper material to select for that joint.

I mentioned that we would machine the inside and outside of the joint and also that we have done parametric analyses to determine what the proper thickness of the joints should be in order to minimize the sustained stresses in this joint. This was done so that we don't have a condition where we are aggravating the joint with it also being a weak link. We design it in such a way that the system stresses, that is, the expansion type stresses that come from the piping system, are low. So, the transition joint regions are all thicker normally than any of the piping that attaches to them. So, we are avoiding elastic follow-up or ratchetting problems there.

Some additional information about the weld process is presented in Figure 13. The welding is done with a hot wire gas tungsten arc welding process. This produces minimal dilution because the heat input is low. Any weld defects are minimized, and we can maintain close control on the welding process parameters. The delta ferrite is low in relation to what would normally be obtained with 308 and tends to be on the low side of six-to-ten.

There are no field welds with dissimilar materials. To reiterate and emphasize all the dissimilar metal joints, this ferritic to 800 to 316, are shop welds using the procedure that I just mentioned above, with all the close controls and the machining. Only like metal welds are going to be done in the field; either 2 1/4-to-2 1/4 or austenitic-to-austenitic.

STRUCTURAL EVALUATION METHODS

We have covered criteria and the description of the joints; now (Figure 14) I will discuss the structural evaluation methods. Figure 15 shows what we did. For all the joints, detailed load control analyses were performed to show compliance with all the applicable design criteria, consistent with all the loading conditions specified, and with the proper combination of loads, and so forth. That is one step. This step satisfies load control methods.

Now we consider the deformation and strain control limits. We performed detailed elastic strain control analyses for all of the cold joints. Now, also for the cold joints, because they do see some time at elevated temperatures (a very short time of about 16 hours for the whole 30-year plant operation), we want to be sure that we've captured the full peak strain range that there might be in these joints locally. So, we perform an inelastic analysis. It says it's simplified but it is not all that simplified.

It is a detailed analysis. It is a true inelastic analysis, and it is not a screening rule. This was done to these joints in order to account for a high-temperature effect.

For these joints, as for the hot joints, the stress-free condition was considered to be 1350°F. That is the temperature at which post-weld heat treatment takes place. The joints we are talking about here, the critical joints, are always 2 1/4 chrome to 800. The other joints don't see that same post-weld heat treatment, but also, they are not the critical joints.

For the critical joints, the analysis captures the residual stress situation in a reasonable and justifiable way by starting at the post-weld heat treatment temperature. All these joints do see the Code-required post-weld heat treatments to get the material in the right set of conditions.

So, analysis starts in kind of an unconventional way on these joints and the purpose is to capture the residual stress effects.

The fatigue damage caused by the stress relief itself was accounted for in the analysis. We did inelastic analysis of the hot joint, and the cooldown from 1350°F certainly was part of the total strain range inventory that we considered. The cold leg joint analysis did not specifically account for the cooldown damage but evaluated it qualitatively by examining the total fatigue damage, and it is small. Therefore, the addition of what we would call a half-cycle was considered not to be significant. If the damage values were higher, we would probably want to take a closer look at that.

Figure 16 shows the minimum design margin for load controlled analyses. Looking at all load controlled stresses, it includes seismic loads, sodium water reaction loads, all the total stress inventory for all the joints during the entire service life. The smallest design margin calculated is 22 percent for Joint Numbers 3 and 4. We lumped these together for convenience so that we don't have to do eleven separate analyses. We were able to envelope and group joints together so that we could analyze significant joints and give a worst case calculation. So, these are worst cases here. In the table, the design margin is defined as the allowable stress divided by the actual, minus one.*

If the stresses were at 80 percent of the allowable, that margin would be 0.25. The smallest acceptable margin is zero, which would mean that the design is right up to the allowable. As long as these numbers are positive, we have a condition where the applied stress is lower than the allowable stress. This finishes the load controlled stress evaluations.

The next consideration is the strain controlled analysis. Figure 17 shows just the so-called cold leg joints. We only show two here because, again, we grouped for convenience. Although only transition joint number 2 is shown, it really represents the worst case of 2 and several other joints, and similarly for joint number 8. The calculated accumulated strain is 0.08 percent, and there is no ratchetting.

We now look at the total creep fatigue damage factors for the different materials. The maximum creep fatigue damage factors are about 0.31, with 1.0 being acceptable. It is the maximum acceptable. If the damage factor exceeds 1, it is not acceptable. So, this is a little bit different kind of number than has been shown in the previous figures. Thus, the worst case is about one-third of the allowable.

This shows that the damage factors for Inco 800 are significantly smaller than Cr-Mo, and 316 is also smaller. But you can see that those materials do have some significant amount of fatigue damage and essentially no creep damage, as might be expected. This is for full 30-year life, so the message of this chart is that all the points that are called low-temperature cold leg joints satisfy all of the Code criteria for full 30-year life of plant operation. In the chart, the nomenclature D_c and D_f denote the amount of damage from creep and from fatigue, respectively. The term ϵ_c denotes the amount of

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*design margin = $\frac{\text{Allowable Stress}}{\text{Actual Stress}} - 1$

accumulated creep strain calculated by the simplified inelastic analysis, which is not all that simple; but a bounding worst case kind of calculation to determine how much localized strain can result from the few hours of elevated temperature that we actually expect to experience. In the region of the welds, there are no factors applied just for the fact that there is a metallurgical discontinuity. We did, though, calculate in the 2 1/4 Cr-1 Mo region, which sees the greatest amount of calculated damage, the largest K_e that the low temperature code asks you to calculate and applied the largest of either that factor or the localized calculated inelastic strain from the inelastic analysis. So, whichever was the largest, we applied that to the calculated strain range of fatigue.

That concludes the elastic analyses. Now, inelastic analysis (Figure 18) is used for Joint 1 which is the highest temperature joint and is also the largest joint.

Figure 19 is a flow diagram of the inelastic analysis process applied to this joint. Our analysis used the MARC Computer Program. The input into this program is geometry, element type, shape, load history, and all the detail of the load history that comes out of the design specifications.

All the various material properties and relationships feed into the program. So, subsequently, stepping through the analysis, stresses and strain histories are output. Those get evaluated against some kind of a damage model which also has input to it some more material properties, ending in a life prediction.

Into the loading history we have what amounts to boundary conditions for each joint. All the loadings that come from piping are defined for the boundaries of the spoolpiece. There are boundary conditions there, all defined.

Now, it is the piping analyst's responsibility when he does his analysis to assure us that he does not calculate any loads that are more severe than the ones we use. That way, we bound the problem. The loads used are part of the equipment specification. Otherwise, especially when doing inelastic analysis, every time a snubber is changed, we would have to redo the analysis because we would have new loads.

An important part of the whole loading scenario is the histogram (Figure 20), which is shown schematically. The numbers along the horizontal scale are time steps that were used for the computer analysis and have no other significance. As was mentioned earlier, the important point here is that the post-weld heat-treat temperature is our zero time point, our zero stress point.

The first cycle that the joint sees is a slow cool down from the one hour hold at the post-weld heat treatment temperature. The worst stresses that the joint ever experiences occur at this time. In fact, plastic strains occur during the cool down. So, the stress range history is tracked from this point. The post-weld heat treatment temperature is held for one hour.

We have to simplify the equipment specification's total histogram into something that can be analyzed. So, we envelope, again, transients with other transients. This is standard procedure. In this way, we are always analyzing a transient that is as bad or worse than the transients that are in the equipment specifications, and the order of things is done in a way so that we calculate the maximum amount of damage. This is a conservative process all the way through. On the chart, 1U, 2U, 5U, etc., are all some particular transient having to do with some kind of event. The "U" designates an upset event, and the "N" a normal event. OBE is an upset. These transients and the combinations of events are all consistent with Appendix A of PSAR Section 3.7. You will also find all of these transients identified with their numbers in the latter document. I think you will find the order of the "U" and the "5", for example, turned around so that it is "U5" rather than "5U" in Appendix B of the PSAR.

Figure 21 represents the total finite element model that is put into the MARC program. Just for a little more detail, the region of most interest is blown up and is shown in greater detail.

We used the MARC Program Element 28, which is an isoparametric axisymmetric element with nine points of integration. Figure 22 presents some typical results. This is representative of some point in the analytical process and these are inelastic creep strains.

The shading in the figure is intended to give some feel for where the maximum strains are. The little portion at the upper left is the maximum strain, corresponding roughly to number nine. You see "diminishing strain" toward the lower right. This is in the 2 1/4 Cr-1 Mo steel which we know from analysis and experience is the region of most concern and where we expect the greatest calculated damage. Therefore this discussion will focus on that region. The other regions obviously have stresses and strains, but they are not our major concern. You can see the general pattern of strains in that interface region. And you can see that there is an accumulation right along the interface.

Figure 23 shows stresses in the same region. The outside diameter is on the left. There is one little shaded corner that is the point of maximum effective stress. All the stresses are calculated by the program and we have an extrapolation routine that takes us to the surface. From the integration points, all the stresses are tracked and properly combined and the greatest effective stress is calculated right in that region. Isostress lines are shown. The maximum stresses are at the outside and inside corners with the greatest being on the outside.

In the second figure of this presentation there was a conclusion stated that the joints fail where the greatest damage is calculated. This result is confirmed by testing results and experience.

The joint analyzed is Joint 1, the hot joint, 26 inch OD, one-inch thick. Only inelastic analysis was performed on this joint because it experiences the highest temperature and creep effects were expected to be significant, and they are. This is the worst joint and we feel by doing this analysis, we get a good handle on the worst case.

Pipe bending is conservatively put in as an axial load. We looked at the stresses to determine whether tension or compression is most significant, and it turned out that tension is.

The chart in Figure 24 shows the bottom line of the inelastic analysis of Joint Number 1. Shown here are the calculated values for strain and damage factors, fifteen-year operation, nominal 936°F which is the normal operation temperature. We calculated 0.4 percent average strain. That compares with the limit of half-percent strain mentioned earlier. These are one half of the base metal allowables, so even though the stresses and strains are in base metal, the weld limits are used giving 0.8 percent for linearized stress compared to 1.0; 0.9 compared to 2.5. The average and linearized stresses are really controlling. Shown here is the damage factor which is the comparison against creep and fatigue damage. Most of the damage is creep; a small portion is fatigue, but it doesn't turn out to be controlling. It is only about 35 percent of allowable. And so we see, again, the analysis we did shows satisfactory conditions for 15-year operation.

We put half of the thermal transients into 15 years. Because there are so many transients, we did not run the program for all of them. The program was run to determine how much creep and fatigue damage was accumulated after a few cycles of a particular transient and that answer was multiplied by the number of transients. This is a conventional way to do inelastic analysis and it should produce a conservative result. Usually, the strain ranges are highest at the first few cycles. By doing a few cycles and extrapolating, it should be conservative.

The strains are total inelastic strains. If only creep strains were used, possibly 30 year operation could be attained. In fact, we want to look more carefully at this. The current answer is that the 15 year figure looks good. We're not saying 30 years is not good yet because we want to be careful and look at this in a greater detail. If we can calculate satisfactory situations for 30 years, we certainly would want to do that.

It is thought that, if a crack did form in this weld, it would tend to run right along the interface, between the base metal and the weld. That is consistent with the stress picture. It is assumed that there is no damage from the welding process itself, but all damage is calculated that is incurred from the post-weld decrease in temperature and subsequent to that. One could take one cycle and calculate the strain equal to the thermal expansitivity of the metal from the melting temperatures down to room temperature, and then estimate the fatigue damage.

Relative to the appropriateness of the Code limits, it probably takes more study to determine whether or not the strain limits used could be larger and still provide adequate margins against failure.

With respect to what would be done in the plant after 15 years of operation, the answer to that depends on how things develop on down the line, but the project recognizes the need to assure the integrity for however long the service of that joint is. We are doing what we can to see if we can come up with a 30-year joint. We like to think we are going to make that, but if we don't, we'll take appropriate measures to replace it.

With respect to actual serivce, one can check how the plant actually operates compared to the conditions imposed on us by the equipment specifications, which are believed to be worst case kinds of scenarios for the whole life of the plant. Records will be kept of the service of the plant. It is a reasonably well instrumented plant, so we will have pertinent data. We are, of course, still a long way from actually operating the plant and we have not specified exactly what analysis we might do with the records of the plant transients.

With respect to questions about residual stresses, I think these are relieved by the post-weld heat treat process. We believe that is true. You may argue whether it would lead to 100 percent relief, but we think that it will relieve to the point where the residual stresses are very small at the heat treat temperature. From temperature we track the subsequent plastic behavior of the joint.

The accepted material properties were taken primarily from the NSMH which has received extensive peer review. We believe it is very reliable material data. The strain hardening parameters as NSMH defines them for each material were used. There was plastic deformation on cooling from 1350°F down to room temperature. There was plastic straining on cool-down, but subsequently there was little or none, but there was creep.

The next topic to be discussed is the supporting program (Figure 25) and some of the information that has been derived from it.

Three components similar to that shown in Figure 26 were built and were tested at ETEC in Santa Susana, California in a transition joint life test. They utilize the same materials of construction that will be used in the plant joints, although there was one case where a bimetallic joint was used. That is a 2 1/4 Cr-1 Mo welded to 316 directly, using the Inco 82 weld metal. This represents the austenitic weldment, which is of less interest to us because we know that is not the problem area.

Welds 3 and 4 are the welds of most interest, which are the alloy 800 welded to 2-1/4 Cr-1 Mo with Inco 82. Weld 5 is another weld to get us to stainless steel pipe that is subsequently attached in the ETEC test facility, which is a DOE test facility.

Figure 27 shows some of the parameters for the transition joint life test article, and, just for comparison, the corresponding Clinch River component conditions are shown.

As was mentioned before, the plant articles operate at 936°F. The life test was operated at 1100 degrees. Axial stress is approximately 2000 psi in the plant articles and was 7500 in the test article.

The number of severe thermal transients in 15 years would be about 60 in the plant. 35 transients correspond with 2000 hours and is the number of cycles that the joint actually experiences prior to crack initiation.

The objective of the test was to produce cracking, so this was not some kind of endurance test to demonstrate lifetime but to demonstrate that we actually predicted the correct location of maximum stress and damage and understood the failure modes.

The transients at the plant go from 936°F, and typically reduced down, for a fast transient, about 160 degrees. The temperature changed at some rate less than about 4°F per second. The test article conditions were much more severe, 10°F a second and almost 300°F total delta T. Those were the conditions imposed on the test article.

Figure 28 shows some of the results of the test. As mentioned earlier, cracks did occur in the joints. The objective of the test was to achieve failure of these joints and evaluate those failures.

Cracks occurred only in the 2 1/4 Cr-1 Mo portion of the transition joint, consistent with our prediction. They also occurred at the outside surface. That is also consistent. Failure is defined as the first occurrence of a detectable crack. That turns out to be about a five mil deep crack. This is what we define as failure, not total separation, but the first cracking.

So failure of these joints occurred in times in excess of that which would have been permitted by the ASME Code if an ASME Code evaluation of the test itself had been performed.

Part of the objectives of the program was to show that we had a reliable crack detection method by ultrasonic means to locate quantified cracking in the joints, and this was developed and worked very well at room temperature.

In Figure 29 is shown the relationship in time between the transition joint test article and the Clinch River Plant operating time. These are related by ratios of creep strength or creep damage imposed. What is seen here is, from the vertical axis, the transition joint test time in years, the Clinch River

Plant time in years. If we move along this curve, starting at time zero, the first thing you come to is the 15-year point which, to date, is what we show as the acceptable time for the plant article.

Next comes the Code Case 1592 creep damage limit. This is the summation of time, the minimum time to rupture using the k prime factor and that sort of thing. We would come up to a code limit of something less than 2/10ths of a year for the test articles. That same limit is well beyond the plant lifetime for the plant article.

Proceeding at a quarter of a year of the test operation, cracks occurred. This was for transition joint article number 3, which was the one consistent with a detailed inelastic analysis that we did of the transition joint test article itself. A detailed analysis of that joint was done and test article 3 agreed with it quite well. So transition joint cracking first occurred at a quarter of a year, 2000 hours or so.

The tests continued. Crack growth proceeded to about 20 mills in this period of time, about another 400 or 500 hours. One message you can get out of this, if you look at this scale with respect to these points, is some feel for how fast cracks might progress if they ever occurred in the plant article.

The test article was one inch thick, 18 inches in diameter, which is the same thickness as the hot leg plant joints but slightly smaller in diameter. Everything else is the same in the test article as in the plant's joints. The inelastic analysis method was the same, also. However, the two analyses can't be compared directly because, while the analysis process was the same, there were some differences in criteria and assumptions used for prediction of behavior, as shown in Figure 30. For example, the test article criterion was average time to first cracking compared with the plant criteria consistent with 1592.

The strength correlation used for 2 1/4 Cr-1 Mo for the test article was the average observed for the actual heat of the material. Samples from the actual heat of material were sent to the laboratory where creep tests were performed. Those data were used and did fit in quite well with the population of other data that we have, but slightly lower than the average.

That is what was used as a test article strength correlation. For the plant article, we use a minimum stress to rupture as the Code requires. It is negative 1.65 standard deviations from average. That is in the Code case. This is an indication of the conservatism added to the Code calculation for pipe joints.

The stress value used to evaluate the amount of damage that occurs was 100 percent of the calculated stress for the test article and 111 percent for the plant article. This really corresponds to the 0.9 k prime factor.

For strain limits, a strain limit criterion for failure was not used to determine test article failure but one was used for the plant articles. The limiting condition for the test article was the damage summation to one of time versus creep rupture damage. For the plant article, this was the limiting condition used; times to half of a percent strain, as mentioned earlier.

Figure 31 shows some of the same information in a little different way, on a logarithimic plot. At about 800 hours, we come up to a 1592 strain limit for the test article. If we had done a stress analysis of the test article, we would have bumped up against the strain limit here at this time. At about 1460 hours, we would bump up against the 1592 limit for stress rupture that was shown in previous figures. Proceeding, we come to the point when cracking first occurred, between 2000 and 2300 hours.

These results are for test article 3. This has two joints, welds number 3 and 4, which each represent a plant joint. The cracking in both of them occurred at about the same time. There was not a whole lot of difference from one to the other. So you might say we had two test articles.

Cracking had been predicted to occur, based on all those things in the figure, to be out at about 10,000 hours. In this figure is shown the probability distribution. With a standard probability distribution, the time to expect first cracking would correspond to a 50 percent probable time.

The test parameters are set to be sure that joint failure would occur within the test window time, which was about a year, so when we made the prediction, we felt good about being likely that we would actually get failure in the time that we wanted. In fact, we got failure sooner than we had calculated, as shown.

Now, just for information, the minus 1.65 standard deviations occurs just slightly above the point of cracking. This would represent the Code Case minimum values. Minus 3 sigma is essentially a zero probability point, and that is far out on the plot.

So this probability plot gives you some feel for the scatter of the creep rupture property. If you had very little scatter, the whole curve would be pushed together.

Now, this is the relationship of cracking to prediction. This caused us to review our design criteria and correlation to ask why we have this difference. Actually, we get a number of things out of this. We look at this and say the Code Case keeps us out of trouble; and that's good. On the other hand, we missed this by a fair amount, and so we want to look at what reasons might there be to cause the shift.

Now, we're getting to the point about carbon concentration, Figure 32, which shows the estimated carbon concentration at the weld interface of the plant joint. As mentioned before, plants operate at 936°F; but this was estimated for 950°F and for 15 years and is believed to be a worst case. This shows carbon concentration versus distance from the weld interface. Zero represents the place where you have Inco 82 and 2 1/4 Cr-1 Mo steel interfacing.

From the Cr-Mo side, the concentration starts at about 0.1 percent carbon. As one moves along toward the weld interface, the heat-affected zone, there is a decrease in carbon. This is due to carbon migration. There is some carbon migration that takes place during the weld process, during post weld heat treatment and during operation. So the 2 1/4 Cr-1 Mo does reduce in carbon. It tends to saturate at a value which is based on experimental work that has been reported in applied-technology publications.

The carbon goes from the 2 1/4 Cr-1 Mo over to the Inco 82 side for a number of reasons. This peak point is not precise but is based on some measurement. The exact peak is about 0.9 and occurs close to the interface, as shown.

The effects of the localized carbon migration are summarized in Figure 33. The initial carbon content is about 0.1 percent. For the plant after 15 years at 936°F, the minimum carbon content at that interface is expected to fall to about 0.05 percent. For the transition joint test article, it is estimated that the carbon content fell to about 0.05 percent, the same number.

The general trend is for reduced carbon to result in somewhat reduced creep rupture strength. From Oak Ridge studies of creep rupture as a function of carbon content for steel at 1100°F, we found a reduction factor of two and a half. This means that we would expect for the transition joint article to have its life, or time to first cracking, reduced by a factor of two to three. On Figure 31 that would have the effect of moving the peak over close to the observed crack occurrence. The Oak Ridge data at lower temperatures for this carbon content shows a negligible reduction in creep strength.

Therefore, at this point, we believe that there is a negligible effect of the carbon migration on plant joints. This probably does need some further study but that is the picture as we see it now.

It has been asked whether we really know that we're not going to pull all the carbon away from that interface when the welding and the stress relief are done. We don't think we will remove all the carbon from the 2 1/4 Cr-1 Mo. The studies that have been done tend to show a saturation at about 0.05 percent carbon. The reports that we've read talk about there being a continuous replenishing from the bulk of the metal toward the interface, so that this does tend to saturate. It reaches a point where the carbon is either getting replenished as fast as it is leaving, or there is just no more potential to take it from one point to another.

This is what the experimental work has shown us. We think that, for this particular combination, there is no tendency to get the same kind of denuding of 2 1/4 Cr-1 Mo using Inco 82 weld metal as would occur with 2 1/4 Cr-1 Mo welded with 309 or 16-8-2, which tends to suck carbon away from the HAZ zone much more aggressively. The present joint does seem to resist carbon migration and that is one of the key reasons for choosing Inco 82 metal. It has other attributes, but one of the key things is that the carbon just does not diffuse away to the same extent that it would with a different weld metal. The same ORNL reports from which we got this information included the studies of chrome moly material with very low carbon, approximately 0.009; essentially denuded totally. I don't know if you can get lower than that.

To summarize again (Figure 34), all the cold joints passed 30-year life. The hot joints passed readily for 15-year service. The transition joint test indicated that our analytical procedures were getting answers that were confirmed by the test results, and the location of material cracking was accurately predicted.

DESIGN EVALUATIONS OF CRBRP INTERMEDIATE HEAT TRANSPORT SYSTEM TRANSITION JOINTS

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FIGURE 5.0-1

82-151-01

IHTS TRANSITION JOINTS

LOCATIONS

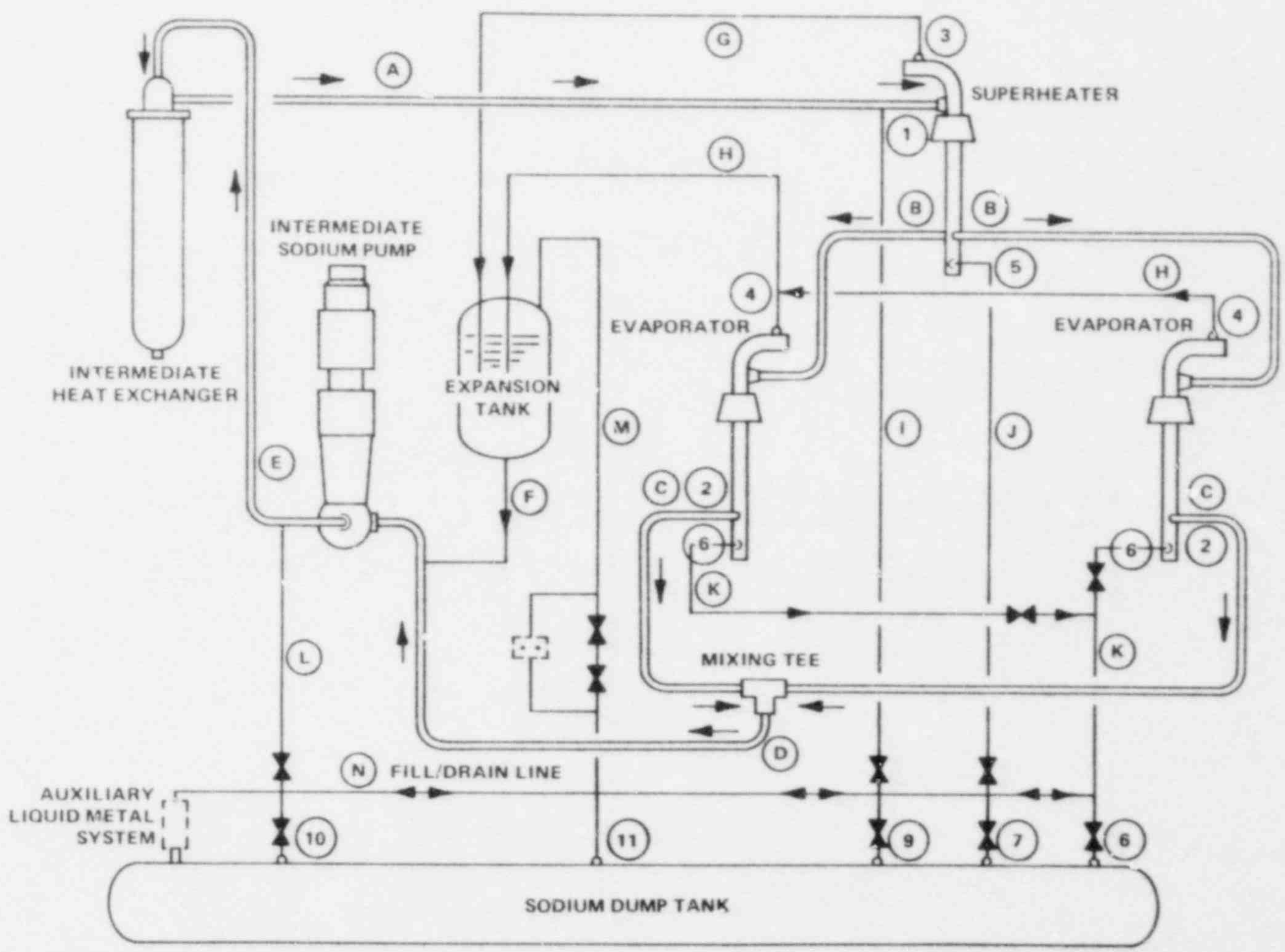


FIGURE 5.0-2

STRUCTURAL EVALUATION

- SUMMARY OF RESULTS OF CRBRP DESIGN EVALUATIONS
- DESIGN CRITERIA
- DESCRIPTION OF TRANSITION JOINTS
- STRUCTURAL EVALUATION METHODS
  - ELASTIC ANALYSIS
  - INELASTIC ANALYSIS
- DESIGN CONFIDENCE DERIVED FROM SUPPORTING PROGRAMS
- CONCLUSIONS

FIGURE 5.0-3

82-151-03

IHTS TRANSITION JOINTS

SUMMARY OF TRANSITION JOINT DESIGN EVALUATIONS

- ALL "COLD" JOINTS PASS ASME CODE ANALYSIS FOR 30 YEAR LIFE

- HOT JOINTS PASS ASME CODE ANALYSIS FOR 15 YEAR LIFE

- TRANSITION JOINT LIFE TEST RESULTS INDICATE PLANT COMPONENT INTEGRITY FOR TIME IN EXCESS OF 15 YEARS

- TRANSITION JOINT LIFE TEST RESULTS CONFIRM THE LOCATION OF THE CRITICAL REGION AS PREDICTED BY ANALYSIS

FIGURE 5.0-4

IHTS TRANSITION JOINTS

117

DESIGN CRITERIA

FIGURE 5.0-5

82-151-05

IHTS TRANSITION JOINTS

1592–7 CRITERIA

- STRAIN AND DEFORMATION LIMITS
  - TOTAL ACCUMULATED INELASTIC STRAINS
    - 1/2, 1, & 2-1/2% RULE
  - CREEP–FATIGUE DAMAGE
    - USAGE FACTOR ≤ 100%

- DESIGN REQUIREMENTS
  - ASME — 1974 SECTION III WITH ADDENDA THRU SUMMER 1975, CLASS 1 CODE CASE 1592-7
  - RDT STANDARDS — E-15-2NB-T SUPPLEMENT SECTION III
    F9-4T SUPPLEMENT CODE CASES 1592-1596
    F2-2 QUALITY ASSURANCE
    F3-6T NDE SUPPLEMENT TO SECTION V
    F6-5T WELDING SUPPLEMENT OF SECTION IX

FIGURE 5.0-6

IHTS TRANSITION JOINTS

DESCRIPTION OF
TRANSITION JOINTS

FIGURE 5.0-7

82-151-07

IHTS TRANSITION JOINTS

SPOOLS

NO.	DESCRIPTION	NOMINAL PIPE SIZE, in.	DESIGN CONDITIONS	NORMAL OPER TEMP	MATERIAL TRANSITION
1	SUPERHEATER INLET	26	325/965	936	2-1/4 Cr-1 Mo TO 316H
2	EVAPORATOR OUTLET	18	325/775	651	2-1/4 Cr-1 Mo TO 304H
3	SUPERHEATER VENT	3	325/965	905	2-1/4 Cr-1 Mo TO 316H
4	EVAPORATOR VENT	3	325/775	626	2-1/4 Cr-1 Mo TO 304H
5	SUPERHEATER DUMP AT SUPERHEATER	6	325/965	650	2-1/4 Cr-1 Mo TO 316H
6	EVAPORATOR DUMP AT EVAPORATOR	6	325/775	650	2-1/4 Cr-1 Mo TO 304H
7	SUPERHEATER DUMP AT DUMP TANK	6	50/800	650	A106B TO 304H
8	EVAPORATOR DUMP AT DUMP TANK	6	50/800	650	A106B TO 304H
9	HOT LEG DUMP AT DUMP TANK	6	50/800	650	A106B TO 304H
10	COLD LEG DUMP AT DUMP TANK	6	50/800	650	A106B TO 304H
11	GAS EQUALIZER LINE AT DUMP TANK	6	50/800	450	A106B TO 304H

FIGURE 5.0-8

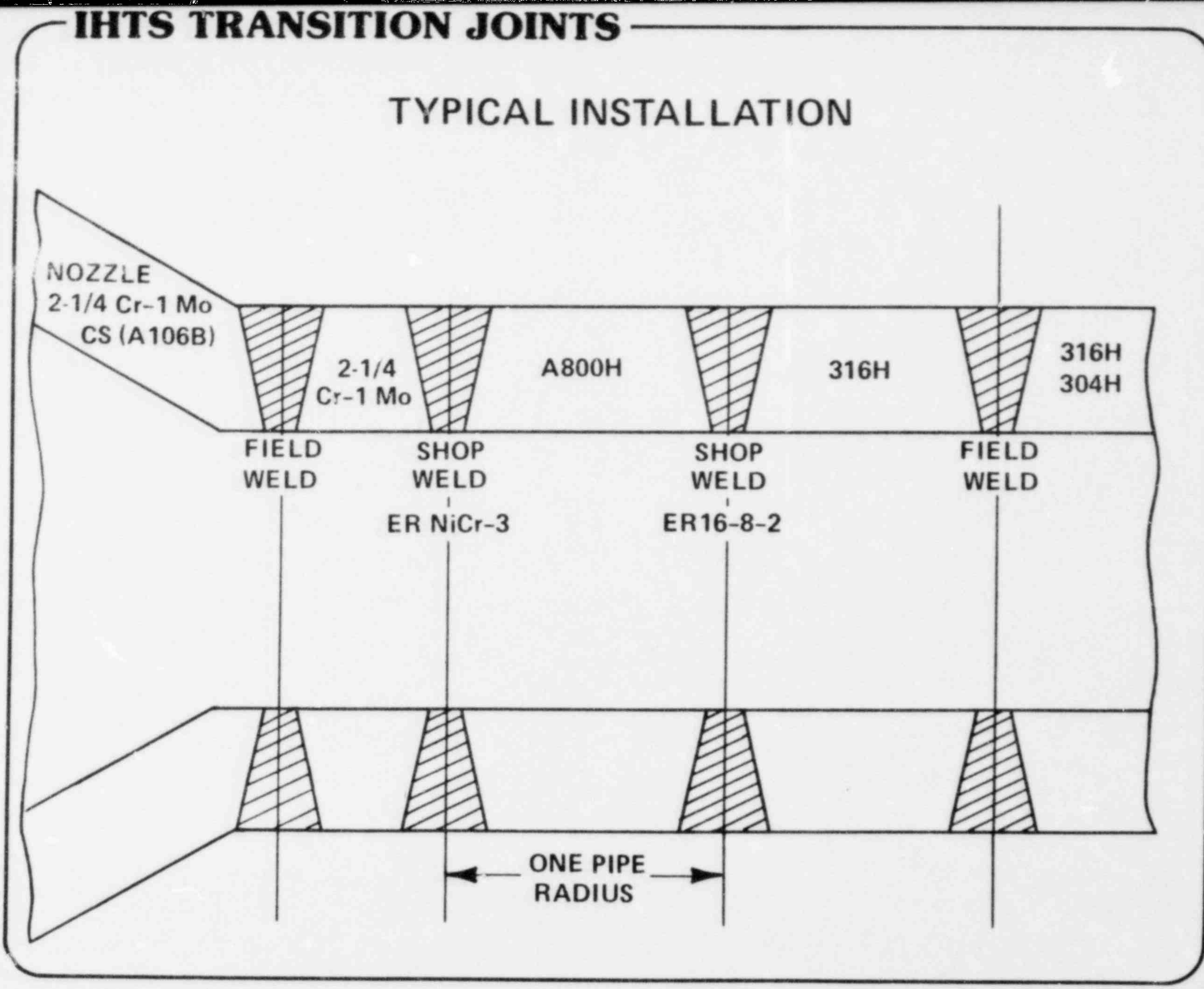


FIGURE 5.0-9

82-151-09

IHTS TRANSITION JOINTS

TRANSITION JOINT WELD DESIGN

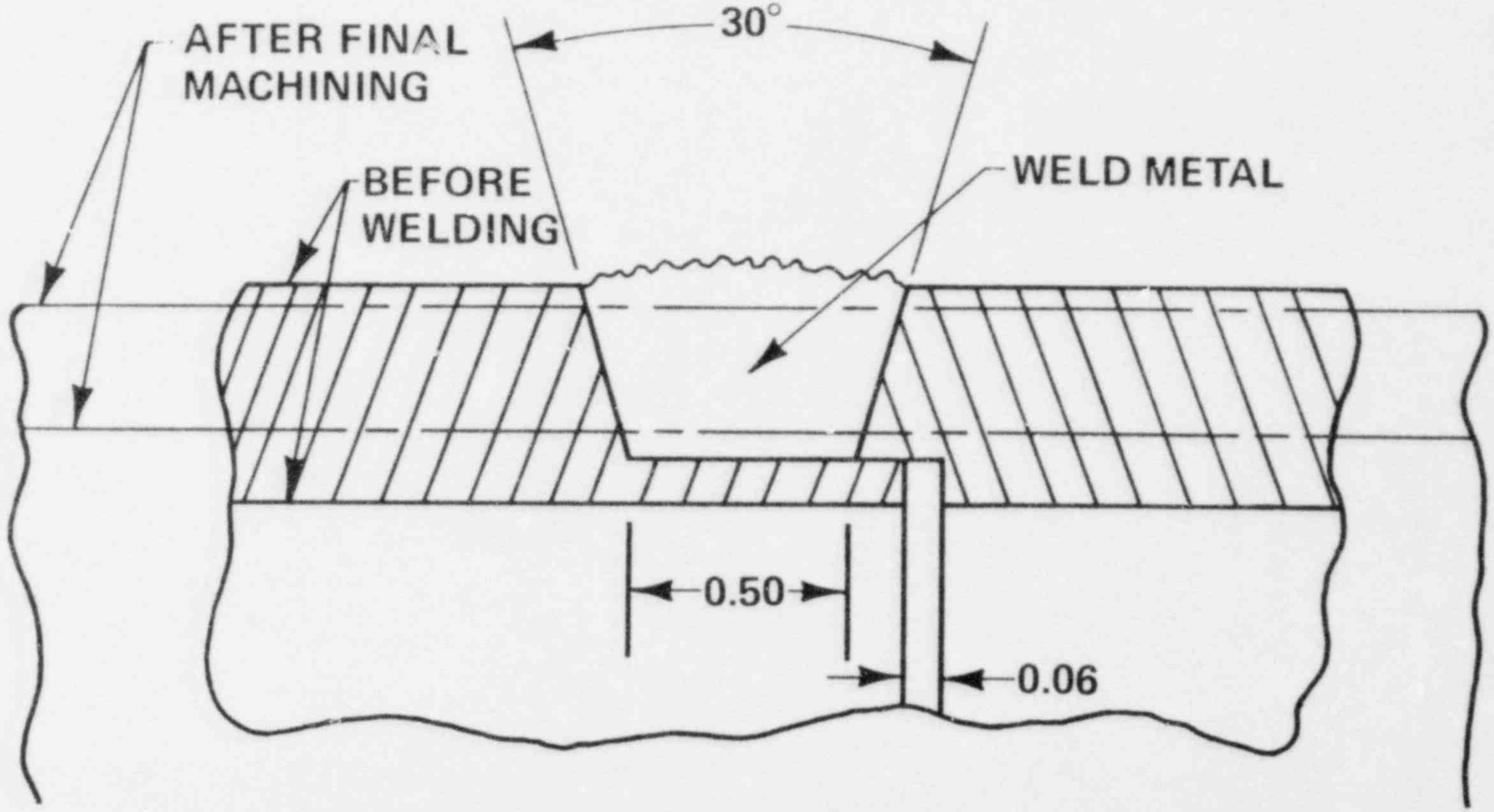


FIGURE 5.0-10

IHTS TRANSITION JOINTS

DESIGN FEATURES

ALLOY 800H TRANSITION MATERIAL

- ASME CODE CASE 1592-7
- INTERMEDIATE COEFFICIENT OF THERMAL EXPANSION BETWEEN 316H AND 2¼Cr-1Mo
- ONE PIPE RADIUS LONG TO ISOLATE WELD STRESSES

ER NiCr-3 WELD METAL

- ASME CODE SPECIFICATION
- INTERMEDIATE COEFFICIENT OF THERMAL EXPANSION BETWEEN 2¼Cr-1Mo AND ALLOY 800H
- HIGH CREEP RUPTURE STRENGTH
- LOW CARBON MIGRATION FROM 2¼Cr-1Mo

FIGURE 5.0-11

82-151-11

IHTS TRANSITION JOINTS

DESIGN FEATURES

ER 16-8-2 WELD METAL

- ASME SPECIFICATION
- INTERMEDIATE COEFFICIENT OF THERMAL EXPANSION BETWEEN A800H AND 316H
- LOW DILUTION AND LOW MICRO FISSURING POTENTIAL
- LOW DELTA FERRITE

SPECIAL THICKNESS STARTING MATERIAL

- INSIDE AND OUTSIDE MACHINE TO ELIMINATE SURFACE STRESSES RESULTING FROM FABRICATION

OPTIMIZED JOINT THICKNESS

- TRADE-OFF OF SYSTEM STRESS VS COMPONENT STRESS

IHTS TRANSITION JOINTS

DESIGN FEATURES

HOT WIRE GAS TUNGSTEN ARC WELDING PROCESS

- MINIMAL DILUTION FROM LOW HEAT INPUT
- MINIMAL WELD DEFECTS
- CLOSE CONTROL OF WELDING PROCESS PARAMETERS

SPECIAL WELD JOINT DESIGN

- WIDE ROOT TO IMPROVE TRANSITION
- WELD ANGLE SELECTED TO MINIMIZE RESIDUAL STRESSES

FIGURE 5.0-13

82-151-13

IHTS TRANSITION JOINTS

STRUCTURAL EVALUATION
METHODS

FIGURE 5.0-14

IHTS TRANSITION JOINTS

ANALYSES TO SATISFY DESIGN REQUIREMENTS

- PERFORM DETAILED LOAD CONTROLLED ANALYSIS TO DEMONSTRATE COMPLIANCE WITH ASME B & PV CODE, CODE CASE 1592-7, RDT-F9-4T, AND E-SPEC, 953089-REV. 22

- PERFORM DETAILED ELASTIC STRAIN CONTROLLED ANALYSES, USING THE ASME B & PV CODE VESSEL RULES (NB-3200)

- PERFORM SIMPLIFIED INELASTIC ANALYSES FOR COLD LEG TRANSITION JOINTS, IN ORDER TO ACCOUNT FOR LIMITED TIME (16 HOURS) AT HIGH TEMPERATURE

- PERFORM DETAILED INELASTIC ANALYSIS OF HIGHEST TEMPERATURE TRANSITION JOINT (T.J. #1) FOR LONG TIME AT HIGH TEMPERATURE

FIGURE 5.0-15

82-151-15

IHTS TRANSITION JOINTS

LOAD CONTROLLED ANALYSIS – SUMMARY

TRANS JOINT NO.	MINIMUM DESIGN MARGINS
1	0.79
2	0.32
3, 4	0.22
5, 6	0.89
7, 8, 9, 10, 11	0.61

FIGURE 5.0-16

IHTS TRANSITION JOINTS

STRAIN CONTROLLED ANALYSIS – SUMMARY

NOMINALLY LOW TEMPERATURE TRANSITION JOINTS						
TRANSITION JOINT NO.	LOCATION	STRAIN EVALUATION	CREEP FATIGUE DAMAGE FACTORS			CODE STATUS
			2-1/4 Cr-1 Mo	Inc 800 H	316SS	
2	EVAPORATOR OUTLET	$\epsilon_c = 0.08\%$ NO RATCHETTING	$D_c = 0.15$ $D_f = 0.16$	$D_c = 0.00$ $D_f = 0.26$	$D_c = 0.00$ $D_f = 0.32$	ALL CODE LIMITS SATISFIED
8	EVAPORATOR DUMP PIPING AT DUMP TANK	$\epsilon_c = 0.08\%$ NO RATCHETTING	$D_c = 0.15$ $D_f = 0.25$	$D_c = 0.00$ $D_f = 0.19$	$D_c = 0.00$ $D_f = 0.35$	ALL CODE LIMITS SATISFIED

FIGURE 5.0-17

82-151-17

IHTS TRANSITION JOINTS

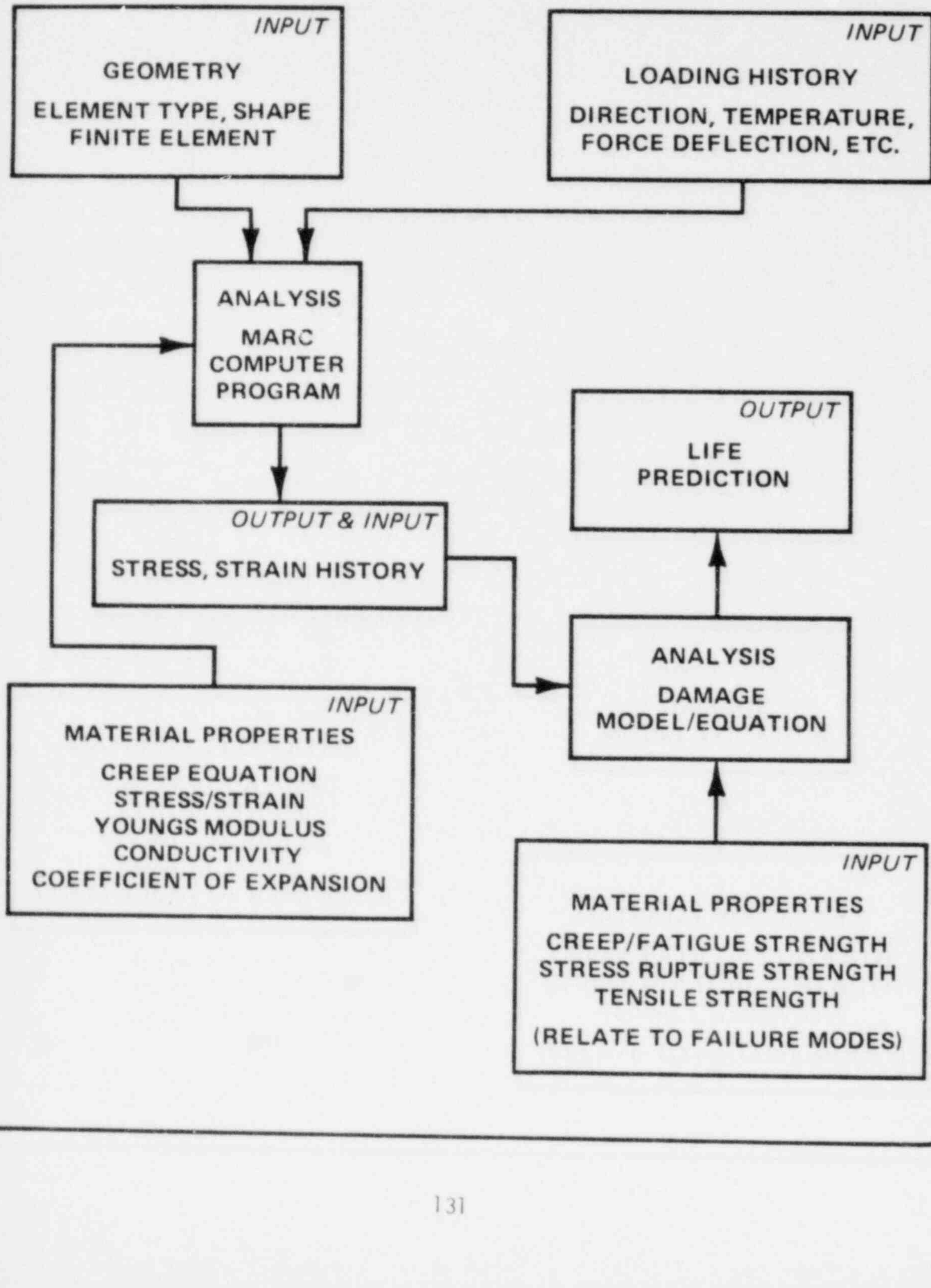
INELASTIC ANALYSIS OF
TRANSITION JOINTS

FIGURE 5.0-18

FIGURE 5.0-19

IHTS TRANSITION JOINTS

ANALYSIS PROCESS

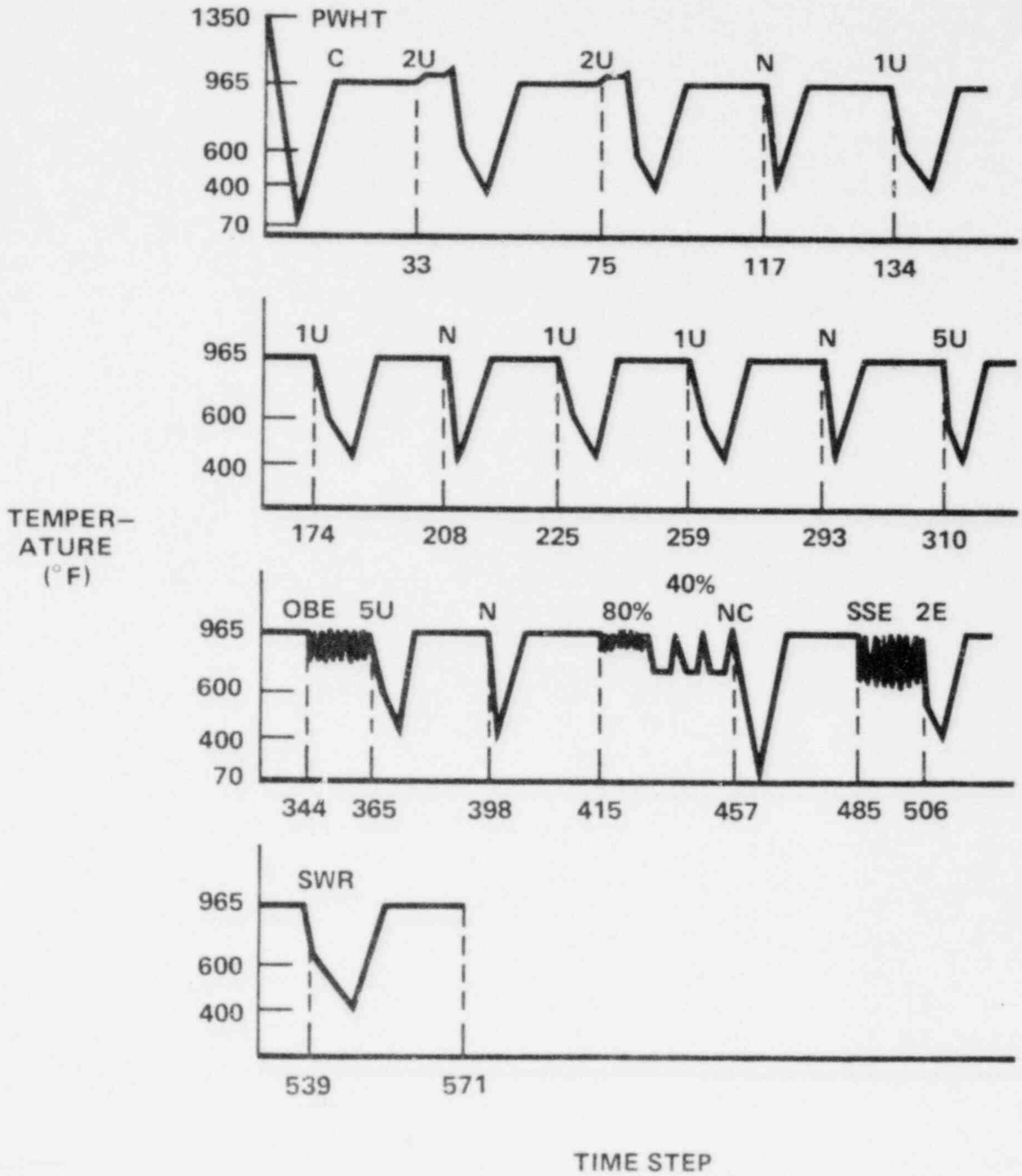


82-151-19

FIGURE 5.0-20

IHTS TRANSITION JOINTS

HISTOGRAM – LOADING SCENARIO



82-15

IHTS TRANSITION JOINTS

FINITE ELEMENT MODEL

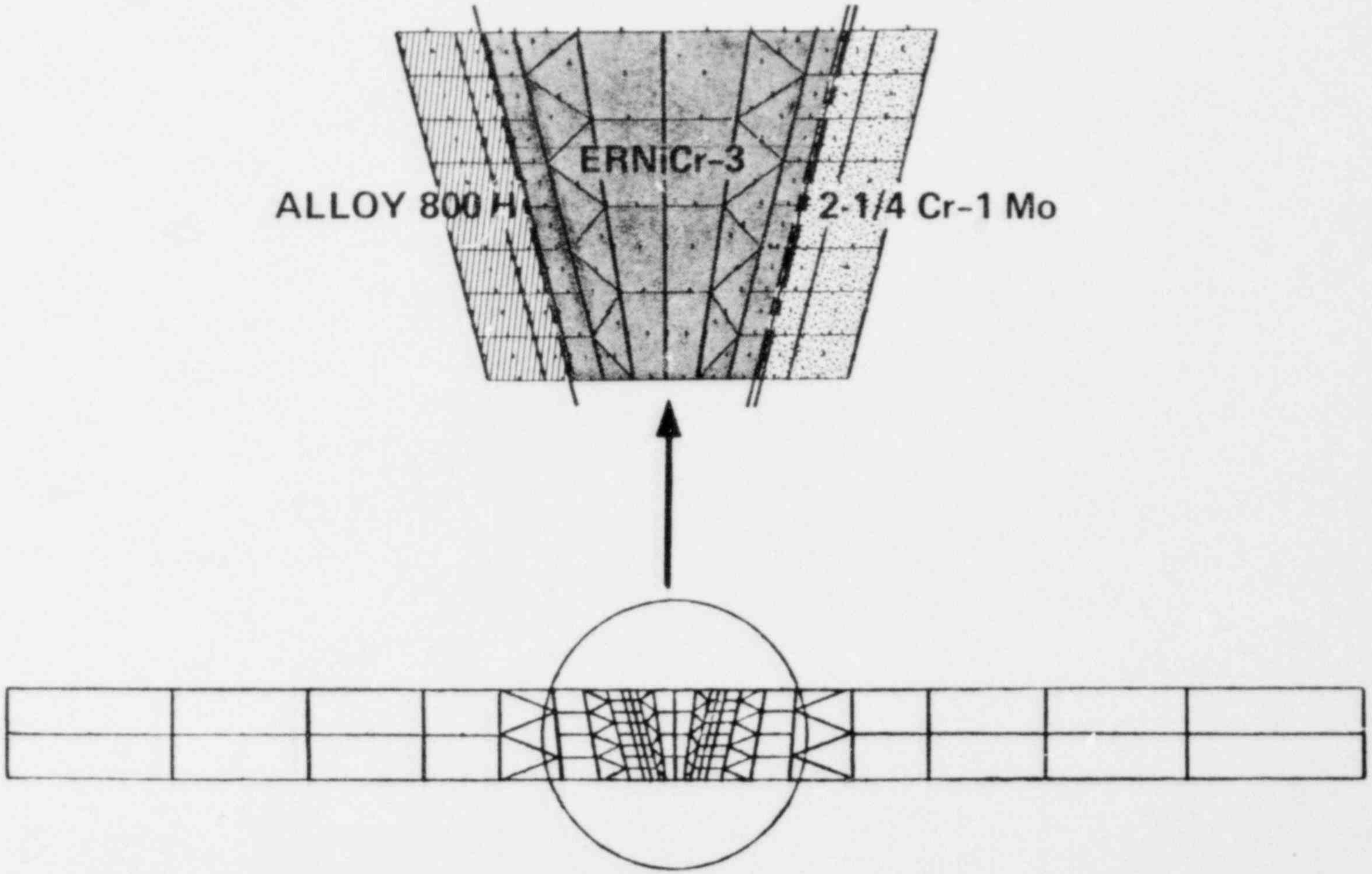


FIGURE 5.0-21

82-151-21

IHTS TRANSITION JOINTS

INELASTIC (CREEP) STRAIN

1 = 0.0043%	2 = 0.013%	3 = 0.0216%	4 = 0.0303%	5 = 0.039%
6 = 0.0476%	7 = 0.0563%	8 = 0.065%	9 = 0.0736%	

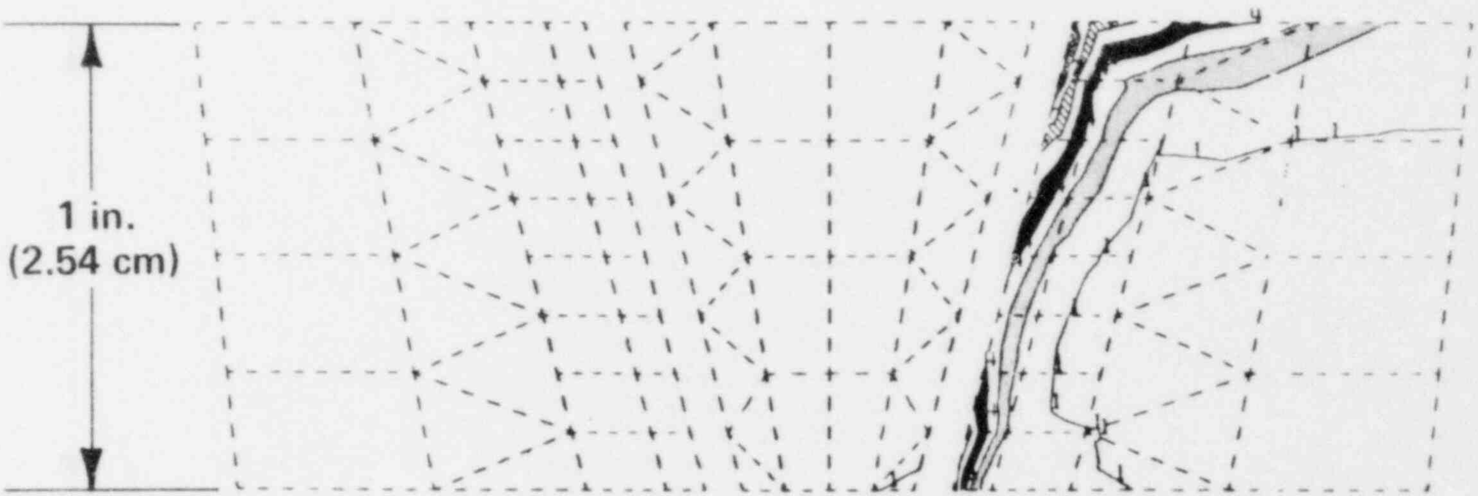


FIGURE 5.0-22

IHTS TRANSITION JOINTS

MISES EFFECTIVE STRESS

1 = 5.79 ksi (38.6 Mpa) 2 = 9.03 ksi (60.2 Mpa) 3 = 12.3 ksi (82 Mpa)
4 = 15.5 ksi (103 Mpa) 5 = 18.8 ksi (125 Mpa) 6 = 22.0 ksi (147 Mpa)
7 = 25.2 ksi (168 Mpa) 8 = 28.5 ksi (190 Mpa) 9 = 31.7 ksi (211 Mpa)

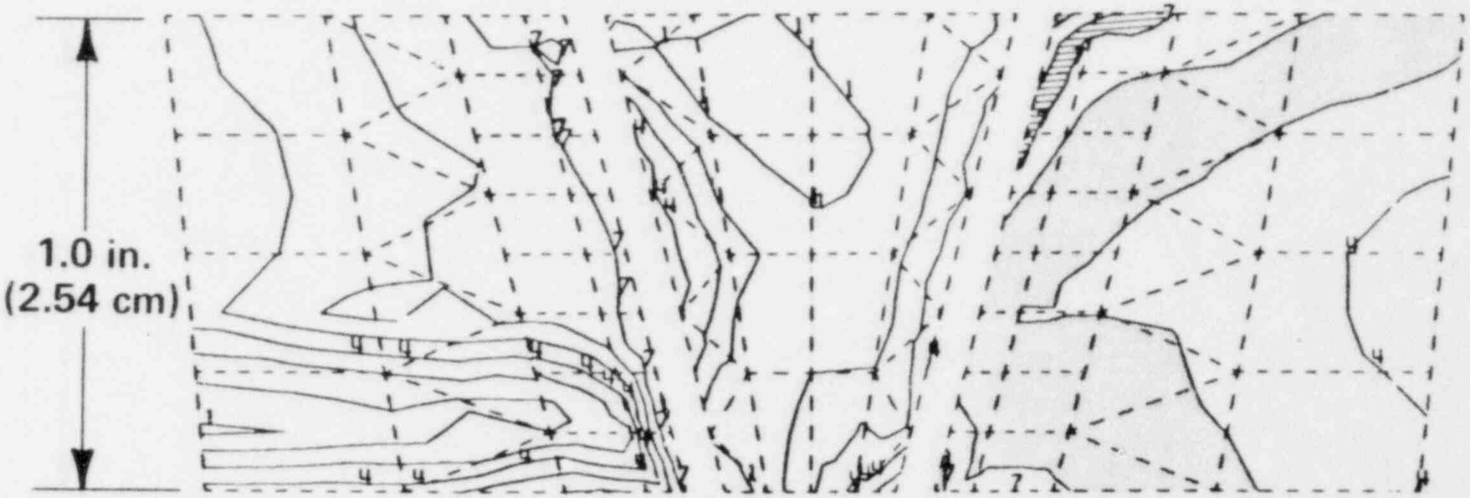


FIGURE 5.0-23

82-151-23

IHTS TRANSITION JOINTS

ESTIMATE OF LIFE FOR 2-1/4 Cr-1 Mo

	CODE LIMIT	15 YEARS (936° F)
ϵ AVERAGE	0.5%	0.4%
ϵ LINEAR SURFACE	1.0%	0.8%
ϵ PEAK	2.5%	0.9%
DAMAGE FACTOR	1.00	0.35

IHTS TRANSITION JOINTS

DESIGN CONFIDENCE DERIVED FROM SUPPORTING PROGRAMS

FIGURE 5.0-25

82-151-25

IHTS TRANSITION JOINTS

SPOOL ASSEMBLY

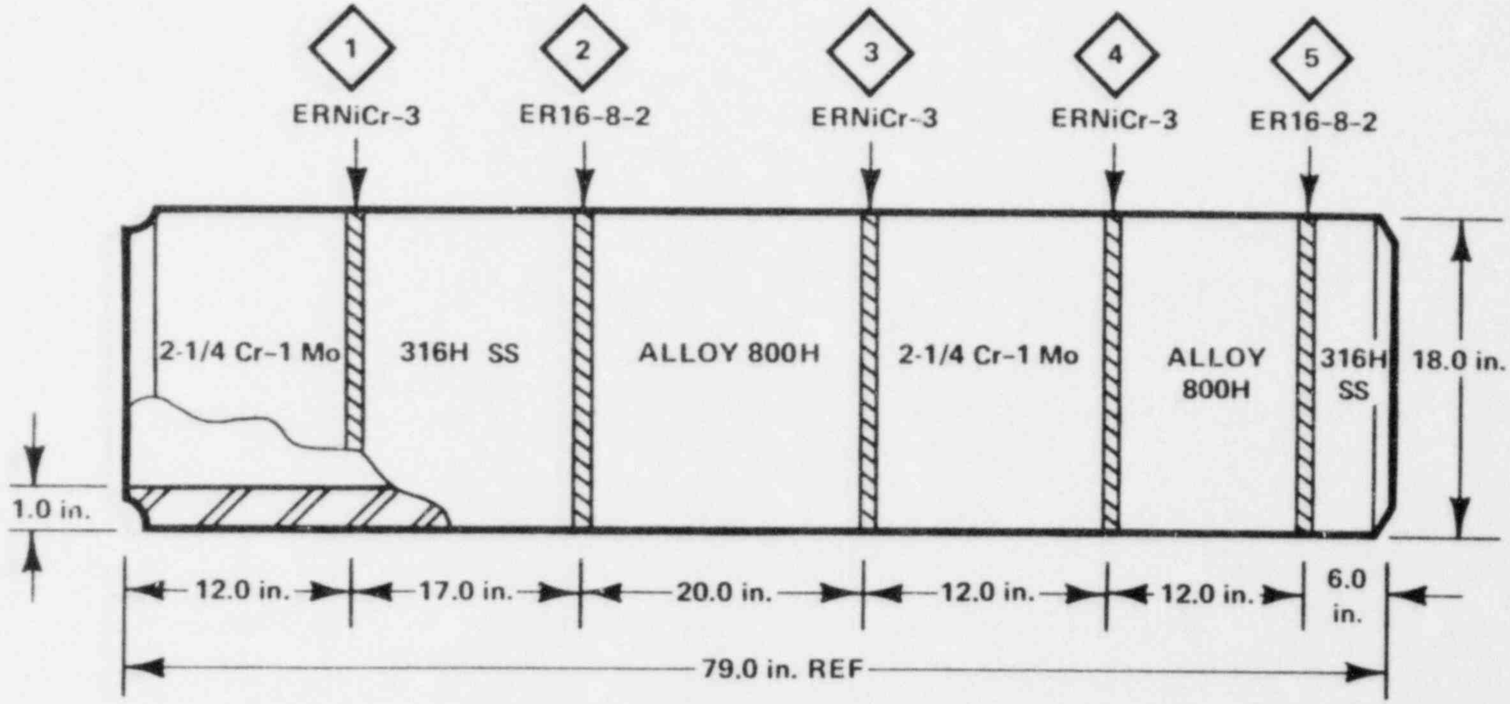


FIGURE 5.0-26

IHTS TRANSITION JOINTS

CRBRP COMPARISON TO TJLT

ITEM	CRBRP	TJLT
LIFE	15 YEARS	2000 HOURS
MAXIMUM TEMPERATURE (°F)	936	1,100
AXIAL STRESS (psi)	2,000	7,500
NUMBER OF TRANSIENTS	60	35
TRANSIENT	936	1,100
ΔT (°F)	160	280
RATE	<4° F/sec	10° F/sec

FIGURE 5.0-27

82-151-27

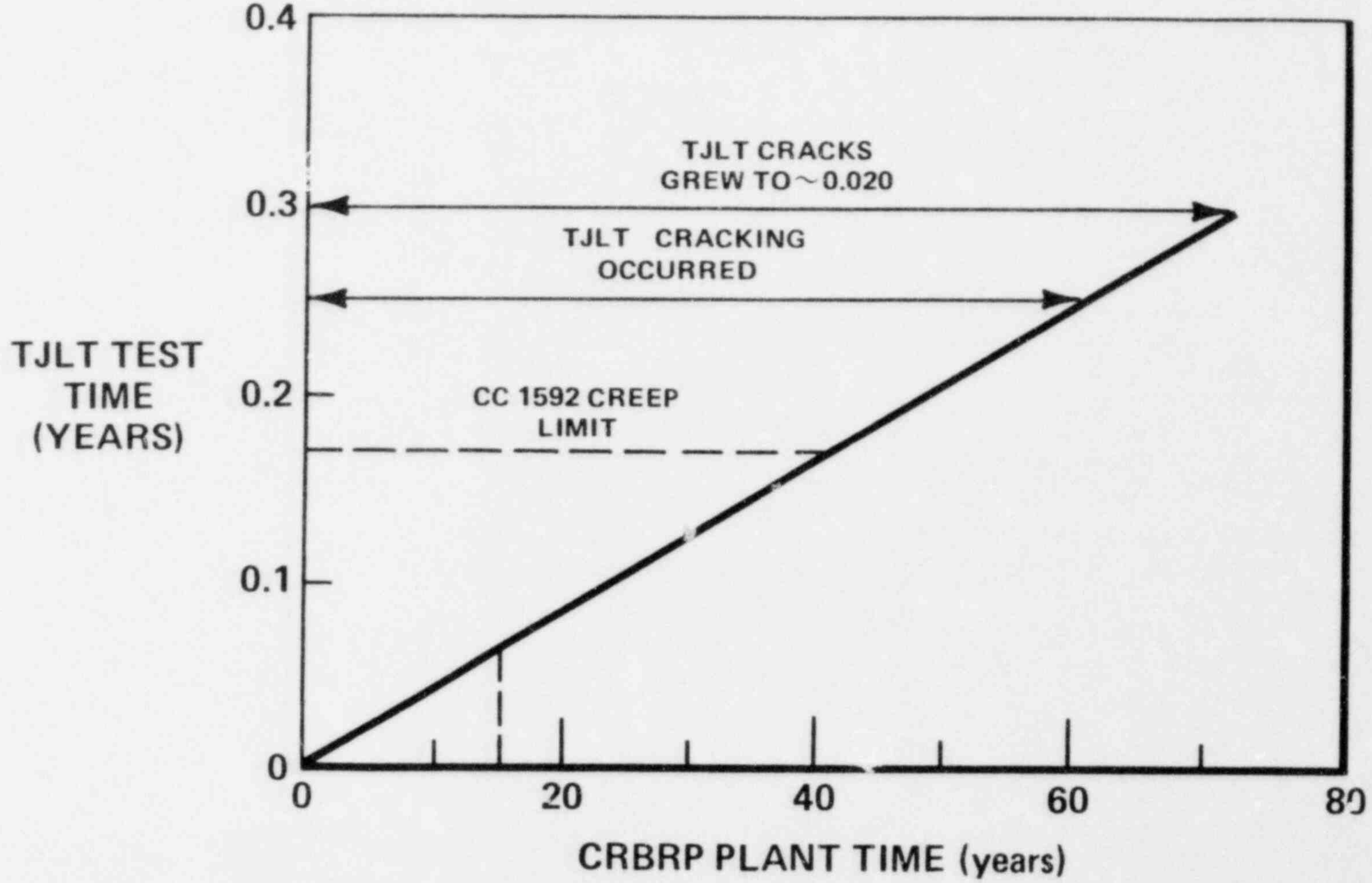
IHTS TRANSITION JOINTS

LIFE TEST SUPPORTIVE RESULTS

- CRACKS OCCURRED ONLY IN CR-MO PORTION OF THE TRANSITION JOINT; CONSISTANT WITH ANALYTICAL PREDICTION
- CRACK INITIATION OCCURRED AT LOCATION OF PEAK PREDICTED MAXIMUM STRESS (OUTSIDE SURFACE)
- TIME TO FAILURE WAS IN EXCESS OF TIME PERMITTED BY ASME CODE DESIGN LIMITS
- ULTRA-SONIC INSPECTION METHOD WAS DEMONSTRATED TO BE CAPABLE OF DETECTING .005 INCH CRACKS IN CRITICAL REGIONS (FERRITIC SIDE OF TRANSITION JOINT) AT ROOM TEMPERATURE

IHTS TRANSITION JOINTS

RELATIONSHIP OF TJLT AND CRBRP TIMES



141

FIGURE 5.0-29

82-151-29

IHTS TRANSITION JOINTS

EVALUATIONS

	CRBRP FOR DETERMINATION OF CODE ACCEPTABILITY	TJLT FOR DETERMINATION OF TIME TO FAILURE
CRITERIA	CC 1592-7 STRAIN LIMIT C/F DAMAGE LIMIT	MINIMUM TIME TO FIRST CRACK
2-1/4 Cr-1 Mo CREEP-RUPTURE STRENGTH CORRELATION	MINIMUM STRESS TO RUPTURE (-1.65σ FROM AVERAGE)	AVERAGE OBSERVED FOR ACTUAL HEAT OF MATERIAL (2 DATA POINTS)
STRESS VALUE USED FOR CREEP DAMAGE EVALUATION	111% OF MAXIMUM CALCULATED EQUIVALENT STRESS	100% OF MAXIMUM CALCULATED EQUIVALENT STRESS
STRAIN LIMIT USED FOR DESIGN EVALUATION	0.5% AVERAGE 1.0% LINEAR BENDING 2.5% LOCAL	NONE
CALCULATED LIMITING CONDITION	TIME TO 0.5% STRAIN	TIME TO $\Sigma \frac{t}{T_R} = 1.0$

IHTS TRANSITION JOINTS

RELATIONSHIPS BETWEEN LIFE PREDICTIONS AND CODE CASE EVALUATIONS FOR THE TRANSITION JOINT LIFE TEST ARTICLE

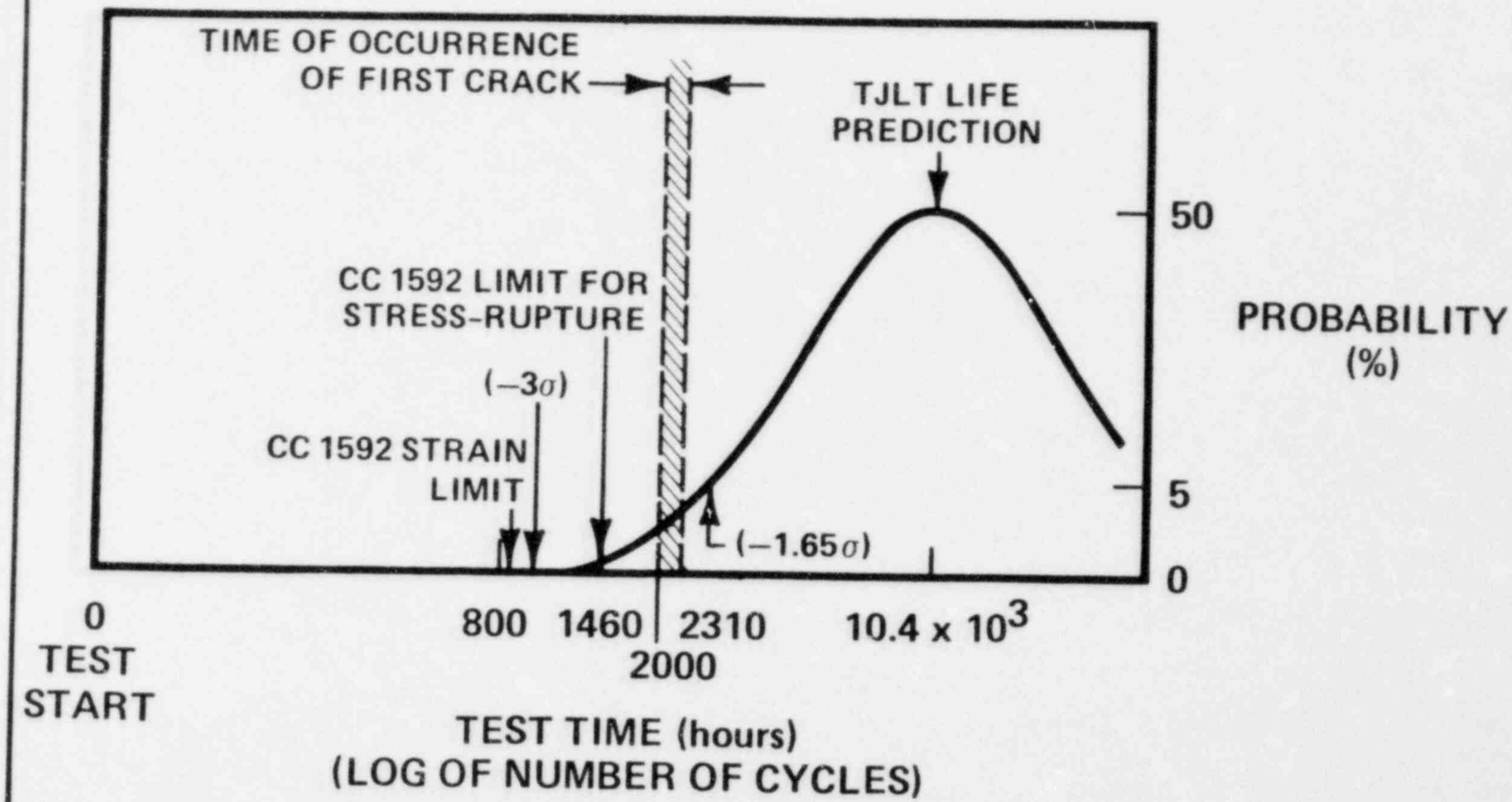
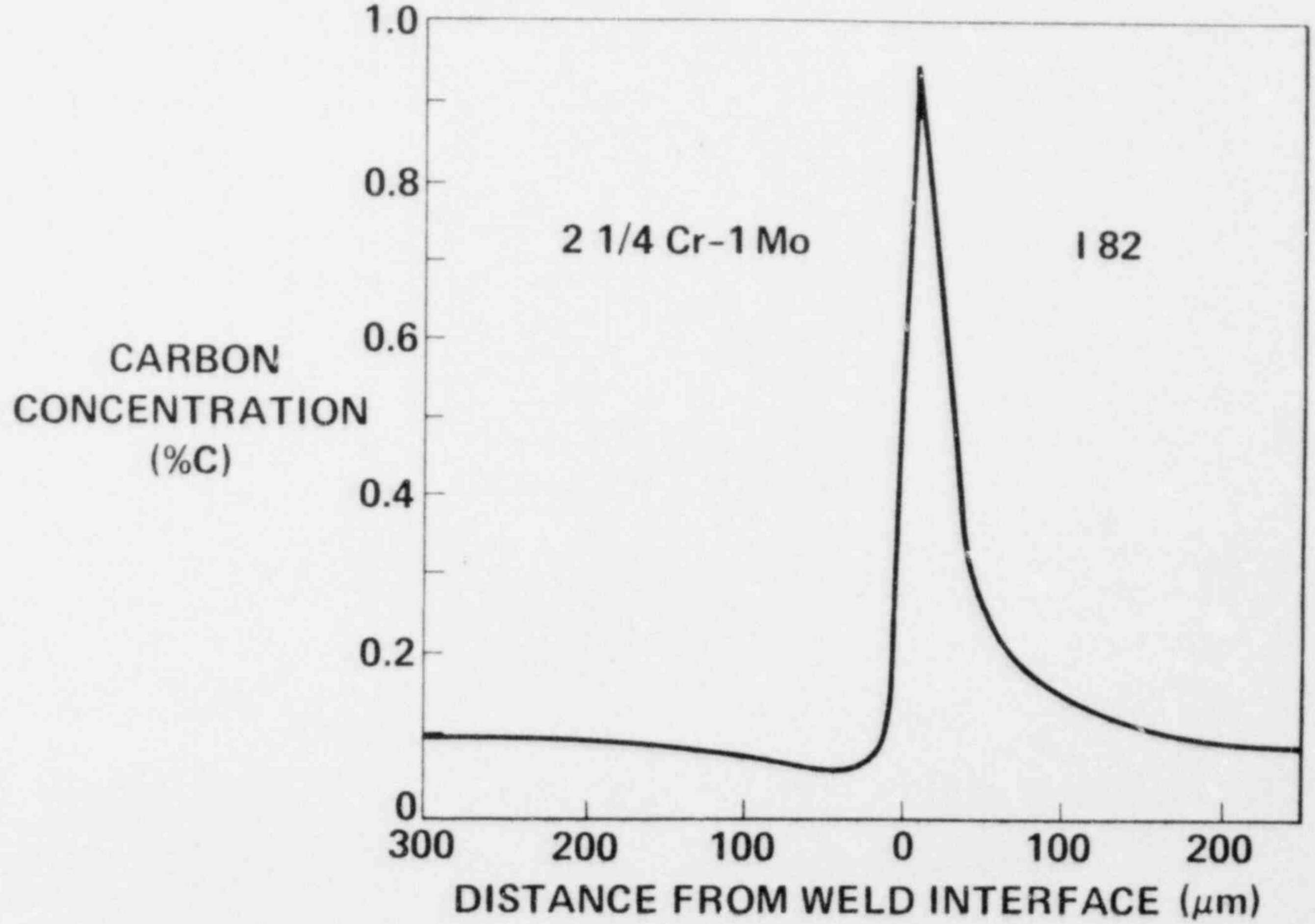


FIGURE 5.0-31

82-151-32

IHTS TRANSITION JOINTS

ESTIMATED CARBON CONCENTRATION AT WELD INTERFACE AT 950° F (15 YEARS)



IHTS TRANSITION JOINTS

EFFECT OF LOCALIZED CARBON MIGRATION

OBSERVATIONS	TRANSITION JOINT ARTICLES	
	CRBRP	TJLT
INITIAL CARBON CONTENT	0.10%	0.10%
CARBON CONTENT AFTER SERVICE	0.05% [1]	0.05% [2]
STRESS-RUPTURE LIFE REDUCTION FACTOR	NEGLIGIBLE [3]	2.5 [3]

1/ 15 YEARS AT 936°F

2/ 2000 HOURS AT 1100°F

3/ BASED ON ORNL STUDIES FOR 2-1/4 Cr-1 Mo STEEL

FIGURE 5.0-33

82-151-34

IHTS TRANSITION JOINTS

SUMMARY OF TRANSITION JOINT DESIGN EVALUATIONS

- ALL "COLD" JOINTS PASS ASME CODE ANALYSIS FOR 30 YEAR LIFE
- HOT JOINTS PASS ASME CODE ANALYSIS FOR 15 YEAR LIFE
- TRANSITION JOINT LIFE TEST RESULTS INDICATE PLANT COMPONENT INTEGRITY FOR TIME IN EXCESS OF 15 YEARS
- TRANSITION JOINT LIFE TEST RESULTS CONFIRM THE LOCATION OF THE CRITICAL REGION AS PREDICTED BY ANALYSIS