NRC FORM 658
(9-1999)

U.S. NUCLEAR REGULATORY COMMISSION

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This form is to be filled out (typed or hand-printed) by the person who announced the meeting (i.e., the person who issued the meeting notice). The completed form, and the attached copy of meeting handout materials, will be sent to the Document Control Desk on the same day of the meeting; under no circumstances will this be done later than the working day after the meeting. ***Do not include proprietary materials.***

DATE OF MEETING: 9/26/2000

The attached document(s), which was/were handed out in this meeting, is/are to be placed in the public domain as soon as possible. The minutes of the meeting will be issued in the near future. Following are administrative details regarding this meeting:

Docket Number(s): 05000247

Plant/Facility Name: Indian Point 2

TAC Number(s) (if available):

Reference Meeting Notice: 00-53

Purpose of Meeting (copy from meeting notice): Regulatory conference to discuss the Apparent Violation and the Risk Significance of the Preliminary Finding, as described in NRC Special Inspection Report 05000247/2000-010.

NAME OF PERSON WHO ISSUED MEETING NOTICE: Peter W. Eselgroth

TITLE: Branch Chief

OFFICE: Region I

DIVISION: Division of Reactor Projects

BRANCH: Projects Branch 2

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Docket File/Central File
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This form was designed using InForms

Consolidated Edison Company Of New York

Indian Point 2
NRC Regulatory Conference

September 26, 2000

OVERVIEW

- Phase 3 Risk Assessment Analysis
- Risk Associated With 2/15 Event
- Risk With Cycle 14 Operation

OVERVIEW

- SG Inspection - 1997
- Measures To Prevent Recurrence

Site Specific Conclusions

- Failed Tube Did Not "Rupture"
  - Actual Leak Rate < 150 gpm
  - Charging Pumps Capacity ~ 225 gpm
- Delta CDF - White
- Delta LERF - Yellow

In-Situ Testing - 2000

- Tested - 51 Tubes, 48 With Indications
  - Tested All Axial Indications (23)
- All Tubes Met 3 Delta-P Burst Margin Criteria
- Negligible Leakage At SLB Test Conditions

ANALYSIS OF PROBABILITY OF RUPTURE

Tom Esselman

6

Objectives

- Identify PWSCC Mechanisms
- IP-2 Crack Growth Is Understood
- Define Likelihood Of Tube Rupture

7

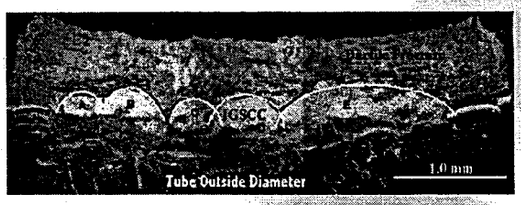
IP2 Row 2 U-Bends With Hour-Glassing

- Stress Distribution
- Mechanism Of PWSCC Crack Initiation And Growth
- Behavior Of The IP-2 Row 2 U-Bends

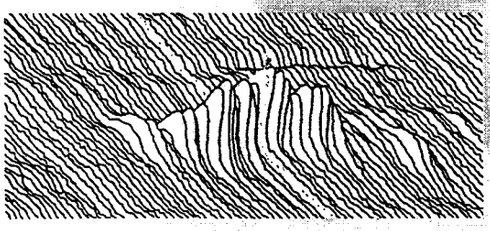
PWSCC Initiation & Growth Process

- Cracks Initiate At Multiple Sites
- Small Cracks Grow And Eventually Link To Form Larger Cracks
- High Aspect Ratio Cracks (Ratio Of Length To Depth) Grow Until Stress in the Remaining Ligament Exceeds Material Failure Stress

Crack Linkup & Growth

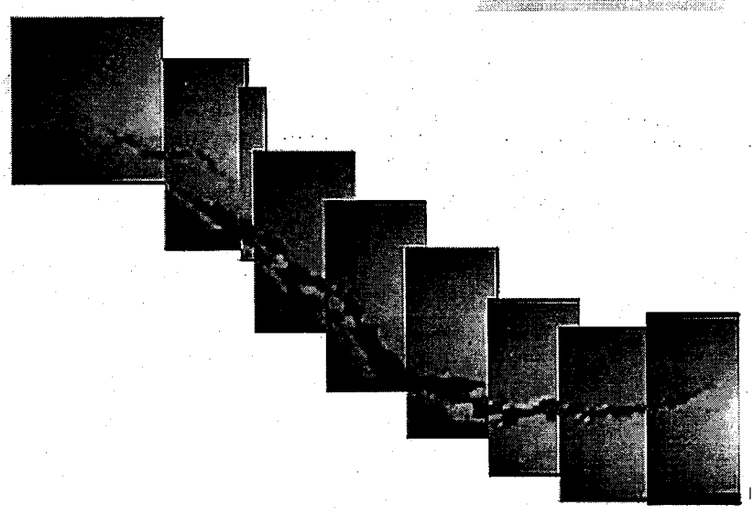


Lab Crack



IP2 SG 24
R2C69

R2C5 CRACK



Mill Annealed Alloy 600 Properties

- **Susceptible To PWSCC**
- **However!!!**

Material is Extremely Ductile

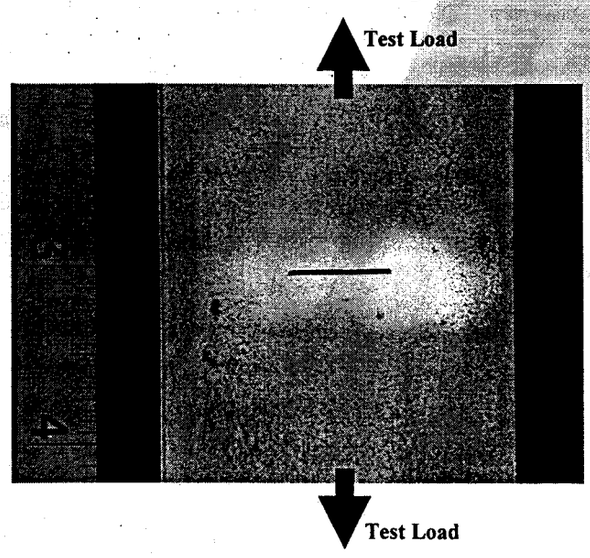
↓

Very High Toughness

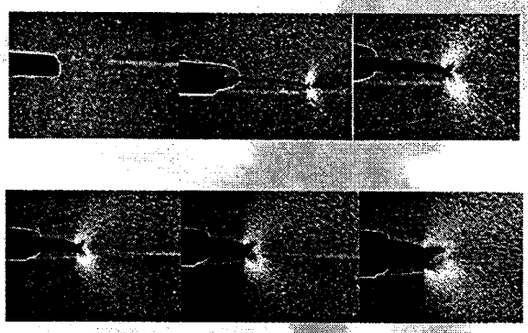
↓

Crack Blunting → Crack Propagation By Overload vs. Unstable Crack Growth

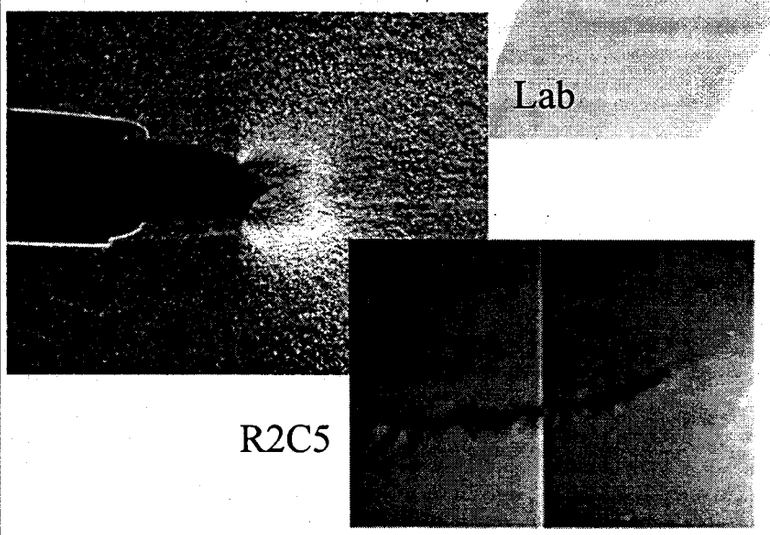
Mill Annealed Alloy 600 Test



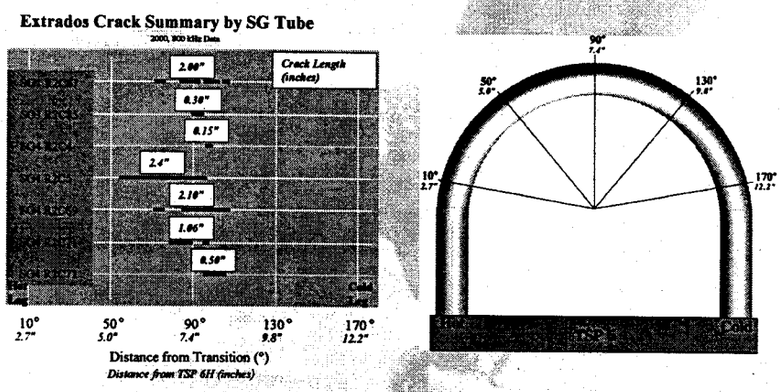
Crack Blunting Behavior



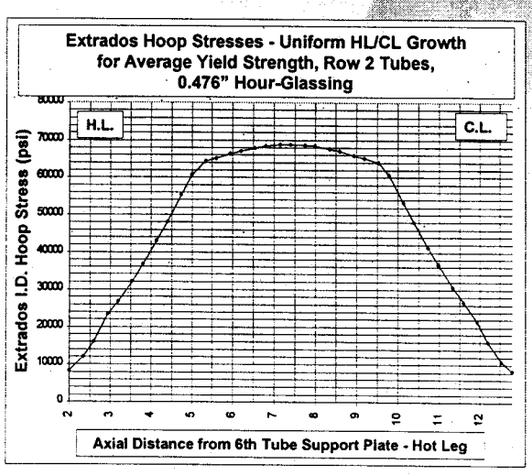
All Loads Above Net Section Yield (49.5 KSI)



Crack Locations



Location And Size Of Cracks Correlate Well With Stress Distribution



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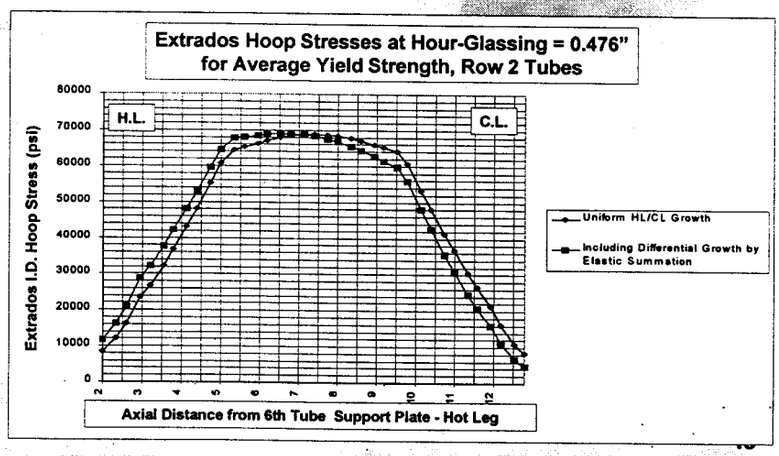

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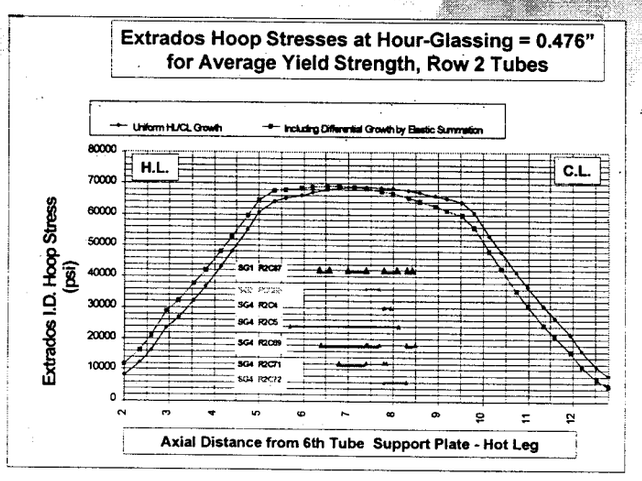

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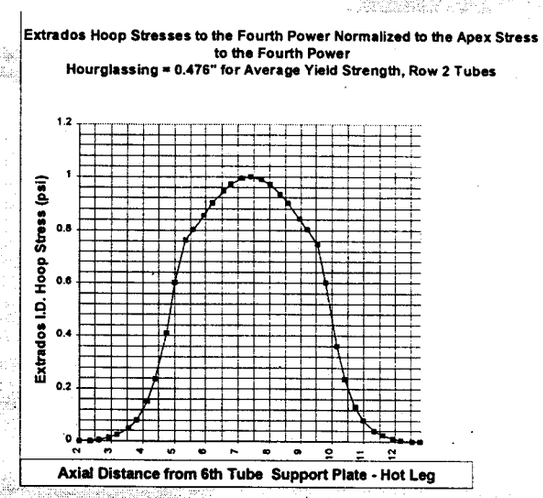


Crack Stability

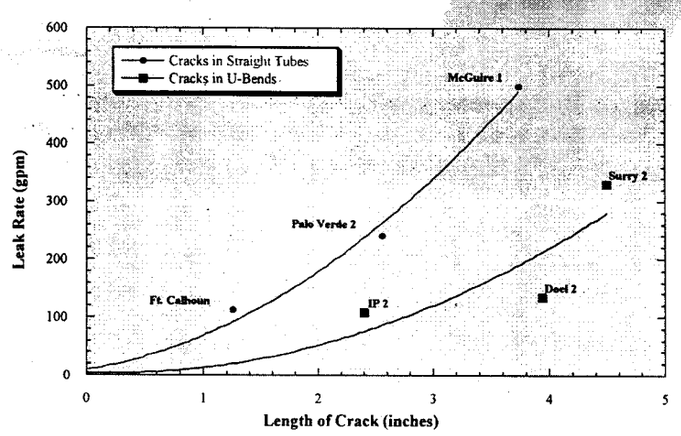
- Cracks Initiate, Grow, And Link
- Linked Cracks Grow Thru-Wall And Then Extend Axially By Linking With Adjacent Cracks
- High Toughness Inhibits Crack Propagation Into Areas With No Cracks Or Shallow Cracks

Correlating Flow Rate With Crack Length

- Equivalent U-Bend Cracks Result In A Smaller Flow Rate, Due To Geometry Restraint And Work Hardened Material.
- Supported By Industry Experience

Leak Rate vs. Crack Length

For Cracks In Straight Tubes And U-Bends



Ref: NUREG 6365

Probability Of Failure
Objectives

- Determine Progressive "Events" Associated With Tube Failure
- Determine Probability Associated With Each Based On IP-2 Conditions
- Use Monte Carlo Analysis Methods To Determine Probability Of Failure

Events

- Number Of Tubes With Undetected Cracks
- Depth Of Cracks
- Crack Growth Rate
- Crack Penetrating Wall
- Axial Length Of Crack
- Flow Rate Through Crack

Postulated Number Of Tubes With Undetected Cracks

- 100 Row 2 U-Bends Assumed To Have Cracks
  - 7 U-Bends Had Axial Indications Identified In 2000 Inspection
  - Conservatively Assumed 100 Tubes Left In Service For Monte Carlo Analysis

Postulated Depth of Cracks

- Assumed A Depth Of Cracks From 0% to 90% Thru-Wall
- Population For Over 50% Thru-Wall Indications Exceeded The Number Found In The 2000 Inspection

Postulated Crack Growth Rate

- Assumed Growth Rates Of 4% To 20% Thru-wall Per EFPY
- 2000 Inspection Showed Growth Rates Of 0% To 16% Thru-wall Per EFPY
- Typical Growth Below 8% Per EFPY

Postulated Crack Penetration Of Wall

- Assumed 100% Probability Of Leakage At 80% Thru-wall
- Assumed 10% Probability Of Leakage At 70% To 80% Thru-wall

Postulated Axial Length Of Crack

- Axial Length Of Cracks Assumed To Range From 0" To 4.5" Long
- Highest Probability Is For Cracks In 2" To 2.5" Range
- 37% Of The Cracks Assumed To Be Longer Than 2.5"
- Assumptions Are Conservative Compared To 2000 Inspection Results

Estimate Of Flow Rates

- Crack Length Converted To Flow Rate Based On NUREG 6365 Data
- If Leakage Occurred, Was It Above Or Below 225 gpm?

Factors for Monte Carlo Analysis

	IP-2	Monte Carlo Analysis
Number of U-Bends With Cracks In 97 Inspection	7	100
Depth of Cracks (# of Tubes Exceeding 50% Thru-wall)	4 Tubes	19 Tubes
Crack Growth Rate (% of Tubes exceeding 8% Thru-wall Per Year)	40%	42%
Axial Length of Flaw (# of Tubes Exceeding 2.5")	0 Tubes	37 Tubes

Monte Carlo Analysis

- 10,000 Trials Performed
- Results Indicated The Following Probabilities:
  - Spontaneous Failure
    - > 225 gpm - .038 Per Year
    - > 75 gpm, < 225 gpm - .275 Per Year
  - Steam Line Break
    - > 225 gpm - .040 Per Demand
    - > 75 gpm, < 225 gpm - .275 Per Demand

Site Specific Risk Assessment

Douglas Gaynor
Con Edison

Site Specific Risk Assessment

Preliminary "RED" Safety Significance Based On Delta CDF And LERF From 4 Postulated Scenarios:

- Spontaneous SGTR
- SGTR Induced By Secondary Depressurization
- SGTR Induced By Over Pressurization
- SGTR Induced By Temperature And Pressure After Core Damage

Site Specific Risk Assessment

SGTR Induced By Over Pressurization

- No Delta CDF From ATWS
- IP2 RPS Model Modified Since IPE
- ATWS CDF For Current Model < 5×10^{-7}

Site Specific Risk Assessment

Temperature Induced SGTR

- IP-2 IPE Used NUREG 1150:
  1.8% Of "High/dry" Sequences Induce SGTR
- Technical Basis Reviewed:
  - IP-2 In-situ Testing
  - 1/7th Scale SG Experiments
  - TMI - 2 Experience
  - Industry Analysis
  - NRC Analysis
- No Change In CDF Or LERF

Site Specific Risk Assessment

Spontaneous SGTR (>225 gpm)

- "Rupture" Frequency of 3.85×10^{-2}/RY
- Evaluated Using IP-2 Full SGTR Model

Site Specific Risk Assessment

Spontaneous SGTR (> 225 gpm)

- Conditional Probability of LERF = 0.13
- Separate Plant Damage States
- Many Sequences Involve Late Releases

Site Specific Risk Assessment

Spontaneous SGTR (<225 gpm)

- Frequency = 2.75×10^{-1}/RY
- Evaluated Using IP-2 SGTR Model
  (adjusted for 225 gpm)

Site Specific Risk Assessment

Spontaneous SGTR

- Results For Two Cases Combined
- Change In CDF = 3.8×10^{-6} / RY
- Change In LERF = 1.1×10^{-6} / RY

Site Specific Risk Assessment

SGTR Induced by Secondary Side Depressurization (SSD)

- Used NRC IE Frequency
- Conditional SG Tube Failure Probabilities
  - 0.28 for > 75 gpm
  - 0.039 for > 225 gpm

Site Specific Risk Assessment

SGTR Induced by Secondary Side Depressurization (SSD)

- Evaluated Using Modified IP-2 SGTR Model
- EOP Guidance and Operator Training

Site Specific Risk Assessment

- IP-2 Emergency Operating Procedures Provide Clear Guidance
  - ECA 3.1 SGTR With Loss of Reactor Coolant - Subcooled Recovery Desired

Site Specific Risk Assessment

- Operator Training on Simulator
  (per crew, lesson every 2 years minimum)
  - Once in 1996
  - Once in 1998
  - Twice in 1999
  - Once in 2000

Site Specific Risk Assessment

SGTR Induced by Secondary Side Depressurization (SSD)

- Evaluated Using Modified IP-2 Risk Assessment Model
- Change in CDF = 2.9×10^{-6}/RY
- Change in LERF = 2.9×10^{-6}/RY

Site Specific Risk Assessment

No.	Postulated Scenario	Con Ed Calculated Δ CDF	Con Ed Calculated Δ LERF
1	ATWS Induced Tube Rupture	0.0	$<5 \times 10^{-7}$
2	High Temp Induced Tube Rupture	0.0	0.0
3	Spontaneous Tube Rupture	3.8×10^{-6}	1.1×10^{-6}
4	Steam Line Break Induced Tube Rupture	2.9×10^{-6}	2.9×10^{-6}
Total		6.7×10^{-6}	$< 4.5 \times 10^{-6}$
Color For Total		WHITE	YELLOW

1997 Inspection
Measures to Prevent Recurrence

J. O. Parry

1997 Inspection

- Meeting July 20th, 2000 with NRC
- 1997 Inspection Performed per Industry Guidelines
  - Scope
  - Probes
  - Oversight
  - Independent Engineering Studies 95, 97

1997 Inspection

Task	EPRI Rev 4	97 Scope
Row 2 & 3	Sample	Inspect 100%
All Tubes	Sample	Inspect 100%
Independent Level III QDA	Not Required	Full Time with Resolution Team
Independent Eng. Evaluation	Not Required	Completed 95 and 97

1997 Inspection Scope and Execution Exceeded then Existing Industry Guidelines

Measures to Prevent Recurrence

- NEI-97-06 Incorporated in Administrative Procedure
  - Required for Outages After 1/1/99
  - Designated SG Project Manager
  - Established SG Management Committee

Measures to Prevent Recurrence

- Degradation Assessment Pre-Outage
- Inspection Enhancements
  - Site Specific Demo for Probes
  - Analyst Performance Tracking

Measures to Prevent Recurrence

- Inspection Enhancements
  - Enhanced Analyst Training
  - 800 kHz probe
  - Improved Data Quality Criteria

Measures to Prevent Recurrence

- Implemented for Inspection of:
  - Original Steam Generators
  - Replacement Steam Generators
- Disseminated Experience to Industry
  - EPRI Seminars (4)
  - Three Assist Visits by Peer Groups

Summary

- Site Specific
  - Delta CDF = White
  - Delta LERF = Yellow